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CHARACTERISTICS OF A SPACE-POWER NUCLEAR REACTOR WITH CONSIDERATIONS FOR VENTING OR CONTAINING GASEOUS FISSION PRODUCTS

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Lewis Research Center
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ABSTRACT

A study of a fast-spectrum, liquid-metal cooled reactor in which the gaseous fission products were either completely vented, completely contained, or partially vented from the fuel element, was performed. Although several areas were found in which insufficient data are currently available to accurately analyze the problem, the characteristic behavior of the various concepts was determined using fuel temperature, cladding strain, and nuclear criticality as limiting conditions. The analysis shows that the completely vented reactor is generally smaller than the other concepts. This size advantage decreases as the strength of the cladding increases and as the allowable fuel temperature decreases.

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SUMMARY

A study of a fast-spectrum, liquid-metal cooled reactor for possible space-power applications was performed to evaluate the feasibility of various fission gas containment concepts. In this study, a ceramic fuel material was used in pin fuel elements. Main objectives of this study were to determine the size requirements of (1) a reactor in which accommodations for venting gaseous fission products had been made and (2) a reactor in which the gaseous fission products were completely retained in the fuel element. Variables considered in the analysis were temperatures, operation times, fuel-pin size and spacing, cladding strength, and the use of different fuels. Limitations imposed in the study included minimum clad thickness, maximum creep allowance in the clad, and allowable fuel temperature.

The results of the study indicated that use of the vented fuel pin generally results in a smaller reactor than the unvented concept but that this advantage decreases as the allowable fuel-temperature decreases and as the strength of the cladding material increases. The study also shows that there is a material strength beyond which only small reductions in core size can be achieved for large increases in material strength.

Another objective of the study was to identify areas in which additional technological information was required to perform an adequate analysis of fuel-element behavior under the conditions imposed by high-temperature, long-term operation. Several such areas were identified, and it was concluded that more experimental and analytical work must be undertaken before a logical choice between the vented and unvented system can be made. Consideration must also be given to some of the potential problems associated with the more complex vented design. These include the difficulties involved with connecting individual vent tubes, loss of fuel from the core, possible plugging of the vent tubes, and possible increase in weight which might be required to shield an external storage tank.

INTRODUCTION

A possible source of energy being considered for future space-power applications is nuclear reactors. Reactors for this purpose must be designed to operate for a substantial period of time with a high degree of reliability. One of the many factors which deter mine the long-life reliability of a nuclear-reactor system is the ability of the fuel elements to maintain structural integrity under the severe conditions imposed by the reactoı environment.

Among the causes of possible fuel-element failure is the accumulation of fission products in the fuel. These fission products may cause swelling of the fuel or, in their gaseous form, can result in a buildup of internal pressure which, in turn, may cause a fuel-element failure. The operating time, power density, fuel temperature, available void volume and the strength of the cladding material determine the magnitude of the problem. Therefore, compact, high-temperature reactors that must operate at a high-power level for an extended period of time are particularly susceptible to this type of failure.

There are two basic approaches to the containment of fission products: the fission products can be constrained within the fuel, or they can be allowed to escape from the fuel into some type of containment volume. For the first of these approaches, a cermet fuel material can be used; individual fuel particles are uniformly distributed within a high-strength metallic matrix which acts to restrain fuel growth. The matrix material also improves the thermal conductivity of the fuel element, enhancing the heat transfer and lowering the operating temperature of the fuel.

A preliminary study of a fast-spectrum, liquid-metal cooled reactor for space-power application was recently completed (ref. 1) in which the size requirements of a cermet core were determined. Because of heat-transfer advantages of triangular spaced coolant holes in a perforated solid matrix, this type of fuel-element geometry was selected over the more generally used rod or pin design. The results of this study (ref. 1) indicated that steady-state heat transfer was not, in itself, a first-order factor in establishing the required core size and the weight of the reactor and biological shield. Nuclea criticality and fuel depletion (burnup) were found to be much more important factors. In particular, with an allowable fuel burnup of 5 atomic percent, the required size of the reactor considered in reference 1 (i.e., 2.5 MW_t, 20 000 hr, L/D = 1) was 10.5 inches (26.7 cm) for a tungsten - uranium 233 dioxide cermet. For a 2-atomic-percent burnup limit, the required core size increased to nearly 18 inches (45.7 cm).

Because the study (ref. 1) also demonstrated that the weight of a 4π-biological shield increased significantly for such a change in core size, the establishment of an allowable fuel-burnup limit for this type of cermet fuel is extremely important. At present, there are very little irradiation data on the behavior of cermet fuels under the conditions required for space-power reactors. Although this lack of data does not rule out the use of cermet fuels, it is impossible to clearly establish an allowable burnup limit for cermets without a substantial amount of in-pile testing.

Rather than trying to accommodate the accumulation of fission products within a cermet matrix, therefore, the alternate approach of permitting the fission products to escape from the fuel might be considered. The gaseous products could then be vented to an external container or, if possible, contained within the fuel element through the use of a strong, high-temperature clad material. For the type of fuel-element design in which the free release of fission products is desirable, a ceramic fuel such as uranium dioxide (UO_2) or uranium nitride (UN) would probably be more effective than a cermet fuel. Likewise, with the use of a ceramic fuel, a pin fuel element could have appreciable fabrication advantages over the perforated fuel-matrix design previously considered (ref. 1). The purpose of this study, therefore, was to investigate the use of both a vented and an unvented fuel-element design to determine the size requirements of a reactor utilizing ceramic fuel pins. A secondary purpose of the study was to identify those areas where a lack of information prevented a complete analysis of the problem. Most of the calculations were performed using a uranium dioxide fuel but some calculations were performed with uranium nitride to show the advantages of using this type of fuel.

In this study, the basic reactor concept presented in reference 1 was used. This reactor was a right circular cylinder with a length-to-diameter ratio (L/D) of 1.0. The core was completely surrounded by a 4-inch (10.2-cm) tungsten reflector. Criticality curves, with an excess reactivity allowance of 10 percent, were established (ref. 1) for various combinations of fuel, tungsten clad, liquid-metal coolant, and void. These criticality limits are used in the present analysis even though cladding materials other than tungsten are considered. The justification for their use is that they represent the most complete set of available parametric criticality curves for reactors of this general description. Preliminary calculations (unpublished work of Wendell Mayo, Lewis Research Center) indicate that, although the use of an alternate cladding material (i.e., other than tungsten) may shift the criticality limits slightly, the shape of the curves will be essentially the same as for the tungsten clad. And, since the purpose of this report is to obtain the qualitative characteristics of the various fuel-pin design, the use of the available tungsten criticality curves was considered acceptable.

A peak-to-average power factor of 1.7 was used in the analysis. This combined factor included a radial power factor of 1.10, an axial factor of 1.20, an allowance for control swing of 1.12, and a hot channel allowance of 1.15.

GENERAL CONSIDERATIONS

For any space-power system, one goal is to minimize the weight of the system. This can be achieved by minimizing the weight of each component in the system. In the case of a nuclear power supply, minimizing core size will usually reduce the reactor weight and the weight of any biological shielding that may be required. The selection of one reactor design over another, however, must also consider reliability, nuclear criticality limits, and fabrication problems. In this study, three basic fuel-element designs were considered. For convenience, these were designated as (1) the vented pin (fig. 1), (2) the

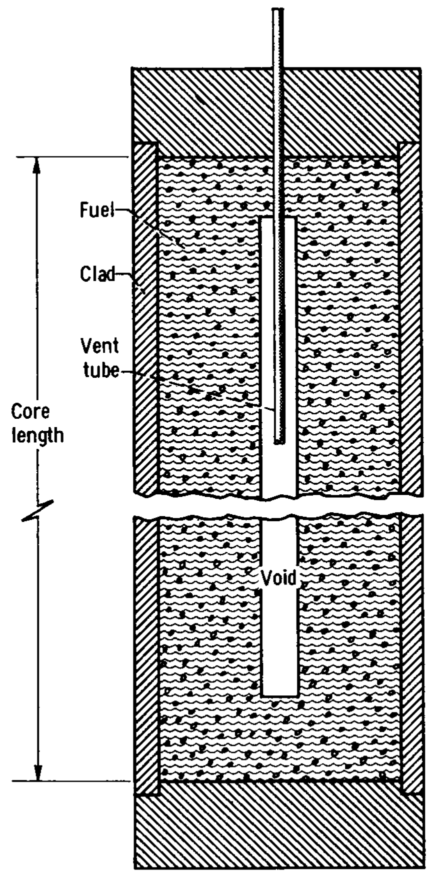


Figure 1. - Vented design.

unvented pin (fig. 2), and (3) the partially vented pin (fig. 3). Each of these fuel-element designs has some advantages and some disadvantages that affect the size and/or complexity of the reactor.

For example, in the first fuel-pin design (fig. 1), the fission gases released from the fuel can be vented to an external storage tank or discharged overboard. In this study,

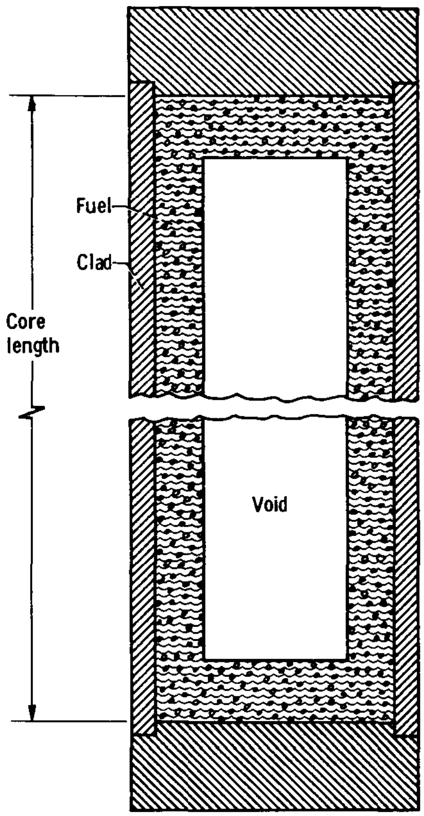


Figure 2. - Unvented design.

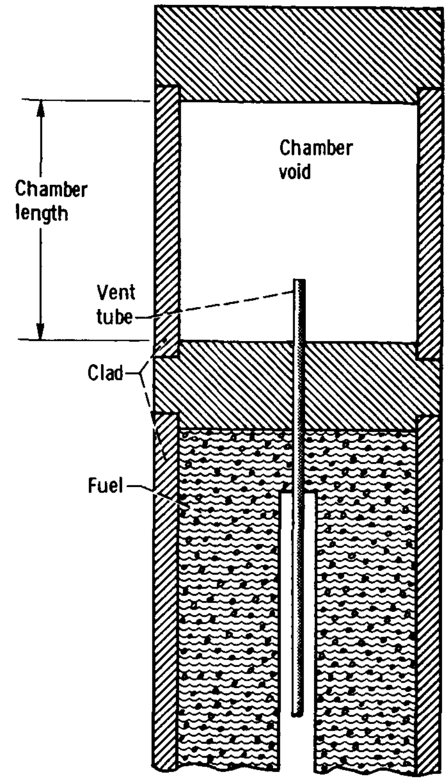


Figure 3. - Partially vented design.

it was assumed that direct overboard discharge would not be permissible for most applications so that an external storage tank would be required. To ensure proper venting, a tube is inserted into a void which has been provided in the central region of the fuel pin. By proper cooling of the external storage container, the resulting gas pressure can be held to a fairly low value even for a relatively small container.

The obvious problems with this concept are (1) the plumbing problem associated with connecting the vent tubes from the individual pins to the external tank, (2) a loss of fuel from the core, (3) a possible plugging of the vent tube by condensation of fuel or fission products, and (4) a possible increase in shielding weight that might be required to shield the external tank.

In the unvented concept (fig. 2), the fission gases are retained within a void space provided in the fuel pin. This results in a simple pin geometry which is relatively easy to fabricate. However, the clad must be capable of withstanding the pressure induced stresses which continue to increase as the fission product inventory increases. Furthermore, after the necessary clad and void volumes have been provided in this concept, there must still be sufficient fuel volume available to meet the nuclear criticality requirements.

The third concept studied is a combination of the vented and unvented pin design. In this concept, the fission gases are vented from the fuel by means of a vent tube, but the storage chamber is an integral part of the fuel pin (fig. 3). Because the external storage tank has been eliminated, the plumbing problem associated with the vented system is also eliminated. However, the loss of fuel from the core and additional shield-weight requirements may still be problems in this design. Limitations imposed by pressure induced stresses in the clad will be similar to that of the unvented design because the minimum temperature at which the gas can be stored, although lower than the temperature of the gas in the unvented design, is still limited by the inlet temperature of the primary coolant. There is also the potential problem of plugging of the vent tube which would aggravate the internal pressure problem. Fuel-element fabrication, although somewhat more complex than the unvented design, should still be relatively simple.

Another factor which must be considered in all three of these concepts is the depletion of the fuel. Assuming an operating life of 50 000 hours, for example, nearly 1.7×10^{25} fissions will occur in a reactor operating at a thermal power level of 3 megawatts. Over 17 pounds (7.7 kg) of uranium will be fissioned under these conditions. If

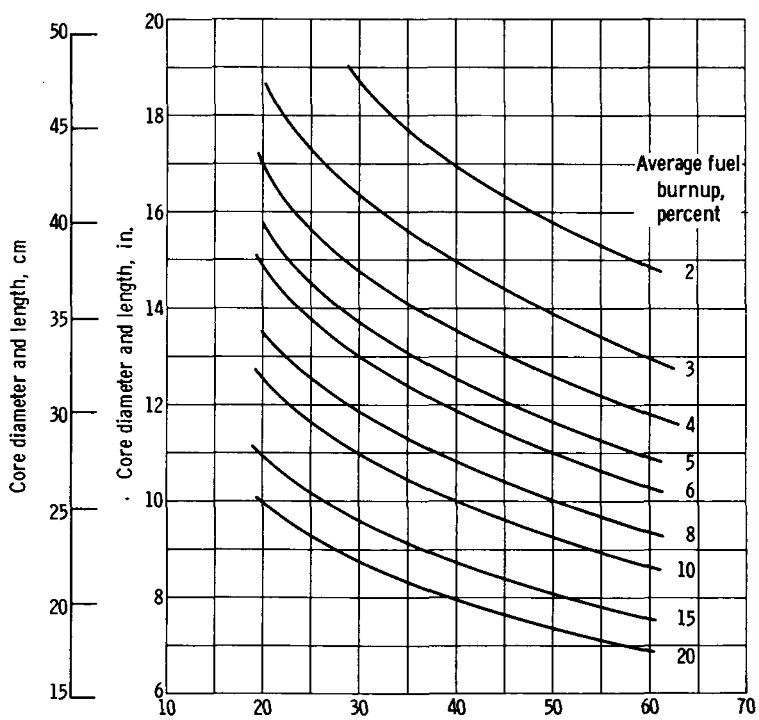


Figure 4. - Effect of available fuel volume and core size on average fuel depletion of uranium 233 nitride in 50 000 hours. Core power, 3 megawatts thermal.

the reactor is small, this fuel depletion represents a relatively large percentage of the available fuel. Figures 4 and 5 show how the average fuel burnup changes with reactor

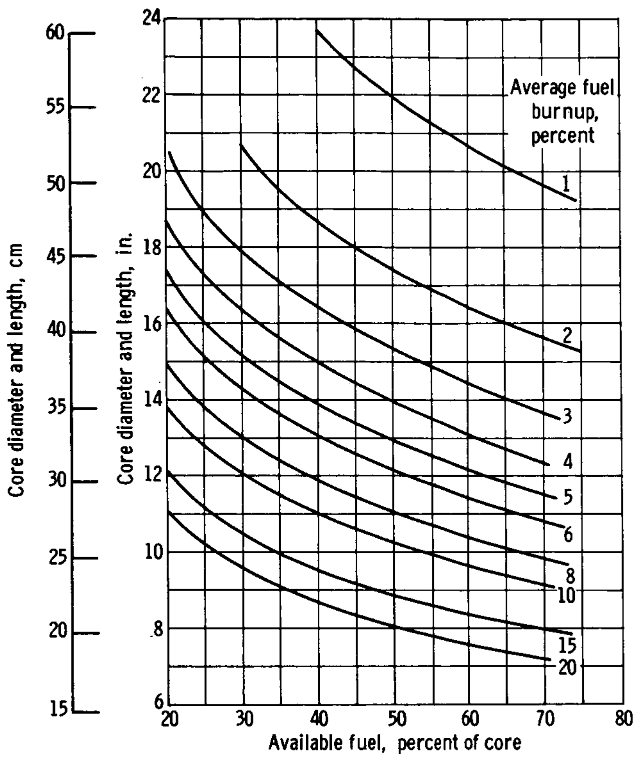


Figure 5. - Effect of available fuel volume and core size on average fuel depletion on uranium 233 dioxide in 50 000 hours. Core power, 3 megawatts thermal.

size and original fuel concentration for, respectively, a UN and a UO_2 fueled core. To limit the fuel burnup to 10 percent, for example, the core size of a UN and UO_2 fueled reactor would have to be greater than 10 and 11 inches (25.4 and 27.9 cm), respectively, at a fuel loading of 40 percent.

BASIC ASSUMPTIONS AND CALCULATIONAL PROCEDURES

Much of the information used in the analysis of the vented and unvented fuel pins is of a preliminary nature. Further work must be done in the areas of the material properties of the cladding and fuel, fission gas release rates, and fuel swelling before a complete analysis of the fuel pin can be performed. However, the purpose of the present study was to determine the characteristics of the various fuel-element designs and to investigate their potential for use in a space-power, nuclear-reactor system. Therefore, the use of the preliminary data should at least yield a qualitative comparison of the various proposed concepts.

Fuel-Pin Model

One of the most important assumptions made was the model of the fuel pin (fig. 6) used in the analysis. This model assumes that the fuel pin is constructed of an outer clad, an annular fuel region, and a central void. The required clad thickness and void volume were determined by optimizing a particular design to obtain the maximum fuel content.

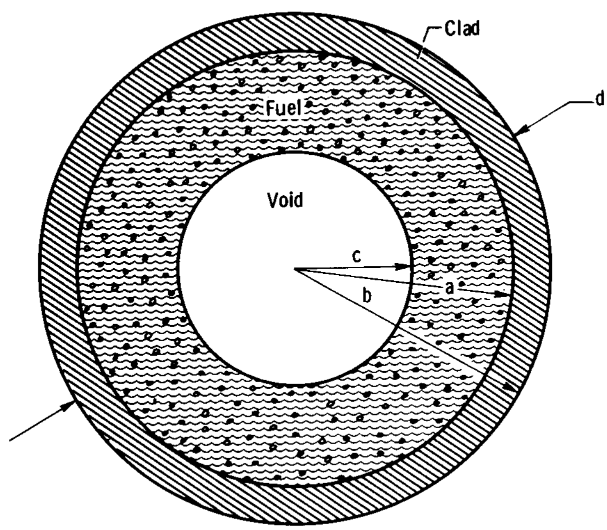


Figure 6. - Model of fuel element used in analysis.

Although the fuel pellets could be initially constructed with the annulus (doughnut) configuration, there is strong evidence (refs. 2 and 3) that, regardless of the fuel form used (e.g., a low density pellet or a loose powder), the oxide fuel would eventually seek the shape assumed in the model (fig. 6). This apparently occurs due to a vaporization-condensation process in which the fuel evaporates from hotter surfaces and condenses on cooler surfaces. The final configuration of the central void (fig. 6) represents a steady-state isotherm on which the condensing and evaporation rates of the fuel are in equilibrium.

Nichols (ref. 4) has derived an analytical method for determining the migration rate of internal voids in oxide fuels as a function of the temperature and temperature gradient in the fuel. This void migration is important to the release of fission products as well as to the central void formation. However, the effect of many of the variables is still unknown and additional work must be performed to accurately establish the time-temperature behavior of the central void formation in ceramic fuel materials.

Because of the mechanism involved in the condensation process, it is assumed that a good thermal bond exists between the fuel and clad (i.e., there is no contact resistance). More experimental work must also be done in this area to validate this assumption.

Fission-Gas Release Rate

Another assumption which was made in the analysis was the amount of fission gas released from the fuel. Although there are some data available on the total fission product yield of U^{233} and U^{235} (refs. 5 and 6) for thermal-spectrum fissions, the information available on the yields of fast-neutron fissions, particularly for U^{233}, is not very complete. For every uranium atom fissioned, there are usually two fission fragments produced for a total yield of approximately 200 percent; yet reference 7, for example, accounts for less than half this amount and only identifies these by mass number rather than by isotopic names. The melting points, boiling points, and vapor pressure of two elements having the same mass number can vary significantly (e.g., xenon 136 and barium 136).

Even if the total fission product yields were known exactly, however, there is still the problem of determining what fraction of the total products produced actually escapes from the fuel and which of these contribute significantly to the pressure buildup. If it were conservatively assumed that all fission products were released from the fuel and all of it became gaseous, the resulting pressure would be extremely high. However, many of the fission products remain within the lattice structure or within microscopic voids in the fuel, so that only a fraction of the total products will be released from the fuel. The products that remain in the fuel may distort the fuel structure and cause swelling, but they do not contribute to the gross gas pressure in the fuel element. Although swelling is a potential problem for high burnup fuels, in this study only the effects of fission gas pressure were considered.

Determination of the gas pressure resulting from the release of a known fraction of fission products is still quite difficult. This complexity is due, in part, to the wide range of melting and boiling points associated with the fission products. Krypton, for example, has a melting point (ref. 8) of 210° R (117 K) and a boiling point of 216° R (120 K) whereas molybdenum (ref. 8) melts at about 5200° R (2890 K) and boils at over 10 000° R (5555 K). There is also a lack of reliable data on the boiling points and vapor pressure of many materials at elevated temperatures and pressures. There is even considerable disagreement on the atmospheric boiling point temperature of some of the more common materials found among the fission products. The boiling point of barium, for example, is given as 2543° R (1413 K) in reference 8 and as 3440° R (1911 K) in reference 9.

Many of the fission products combine with other products making the task of determining the total gas pressure even more difficult. In a uranium dioxide (UO_2) fuel, for example, for each uranium atom fissioned, two atoms of oxygen are released. Several of the fission products, particularly elements such as zirconium, cerium, neodymium, and molybdenum, combine with the free oxygen to form oxides (ref. 10). Other fission products, such as cesium, combine with the bromine and iodine fission products to form bromides and iodides (ref. 10).

Additional work must be performed in the areas involving fission products, release rates, and pressure buildup before an exact determination of the internal pressure of a fuel pin can be made. In the present analysis, it was assumed that, for each fission that occurs, 0.3 atom of gaseous products are released and contribute to the internal pressure of the fuel pin. Although this value (i.e., 0.3 gaseous atom per fission) is somewhat arbitrary, it does include all the fission products which are normally gaseous at room temperature (i.e., xenon and krypton) plus an allowance for the partial pressure of other higher temperature volatiles which might also be released from the fuel (e.g., iodine, bromine, cesium, cerium). A discussion of the influence of this assumption on required reactor size is given in the section, Effect of fission gas release rate (p. 55).

It was also assumed that the fission-product release was linear with time so that the pressure buildup was also a linear function of time. The maximum pressure, which occurs at the end of reactor life, can therefore be related to the number of fissions, available void space, and the temperature of the fuel element.

Clad Stresses and Creep

As a result of the pressure buildup from the gaseous fission products, stresses are induced in the clad. At the temperatures and operating times of interest for space-power systems, these stresses will cause the cladding to slowly distort (creep). It is this distortion which is of concern in the design of reactor fuel elements. For this application, an allowable strain of 1 percent was assumed.

For situations in which creep is caused by the application of a steady stress, there are several empirical methods by which the time necessary to produce either a given strain or to produce failure can be determined. Typical of these methods are those proposed in references 11 to 13. Given the proper experimental data for a particular material, any of these correlations can be used to preduct long-term strains with a moderate degree of success (ref. 14).

When the applied stress varies with time, such as in the case of a linear pressure increase caused by accumulation of fission products, the application of the empirical relations is more difficult. Manson (ref. 15) notes that, although there are still some uncertainties associated with the establishment of an ideal cumulative-creep law, in general, the strains resulting from a series of stress loadings are additive. In the present study, a method proposed by Whitmarsh (ref. 16) has been used to determine the long-term stress-strain behavior of the clad under a linearly increasing load. This method (ref. 16) uses the basic Larson-Miller parameter (ref. 13) except that the stress and strain are integrated over the reactor operating time. The resulting expression for the case of a linear increase in stress is given by

$$\sigma_{max} = B\left(\frac{10^J \tau_{max}}{1 - \frac{1}{mT}}\right)^{mT} \tag{1}$$

where

σ_{max} maximum allowable (end-of-life) stress required to limit total strain to 1 percent

τ_{max} maximum operating time of reactor

T clad temperature

B intercept on semilog Larson-Miller plot for 1-percent strain

m slope of semilog Larson-Miller plot for 1-percent strain

J material constant in Larson-Miller correlation represented by

$$\log \sigma = mT(J + \log \tau) + \log B \tag{2}$$

Figure 7 shows how the allowable (end-of-life) stress (eq. (1)) varies as a function of time and temperature for T-222, a high strength, tantalum base alloy containing 10.4 percent tungsten, 2.4 percent hafnium, and 0.01 percent carbon (ref. 17). The values of the Larson-Miller parameters used in determining the allowable stresses for T-222 (fig. 7) were determined from the data shown in reference 17

$$B = 1.58\times10^7 \text{ psi} \qquad (1.09\times10^7 \text{ N/cm}^2)$$

$$m = -6.58\times10^{-5}/^{\circ}\text{R} \qquad (-1.18\times10^{-4}/\text{K})$$

$$J = 15$$

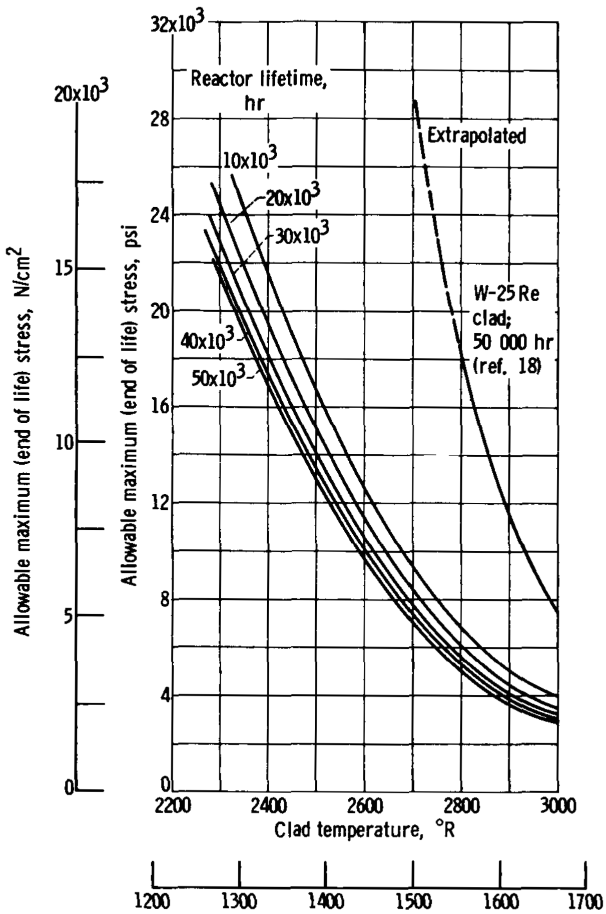


Figure 7. - Allowable maximum stress at end of reactor life for T-222 clad for various clad temperatures.

It can be seen (fig. 7) that the operating temperature of the material has a much stronger direct influence on the allowable stress than the operating time. However, there is another factor, not readily apparent in figure 7, which makes the operating time of the reactor quite important. This additional factor is the increase in fission products and, therefore, the increase in fission gas pressure that occurs with the increasing operating life of the reactor. Since this fission gas pressure directly increases the stress in the cladding (eq. (3)), the longer the operating time of a given reactor, the more difficult it will be to limit the strain in the cladding to a given value. For example, suppose two identical reactors were to be operated for 10 000 and 50 000 hours, respectively. If the reactors had been designed to operate with a clad temperature of 3000° R (1667 K) for 10 000 hours, the maximum allowable clad stress (fig. 7) would be about 4000 psi (2758 N/cm^2). To operate the reactor for 50 000 hours, therefore, the operating temperature of the clad would have to be reduced to about 2400° R (1333 K) to accommodate the greater internal gas pressure and clad stresses resulting from the fivefold increase in fission products.

It is of interest to note how the use of the integrated pressure stress (eq. (1)) changes the design stress of the cladding material. If the internal pressure were assumed to have been constant over the lifetime of the reactor, the allowable stress required to produce 1-percent strain can be determined directly from the Larson-Miller relation (eq. (2)). The ratio of the integrated allowable stress (eq. (1)) to the constantly applied stress (eq. (2)) is then given by

$$\frac{\sigma_{max}}{\sigma} = \left(1 - \frac{1}{mT}\right)^{-mT}$$

Evaluating this expression for T-222 cladding shows that the use of the linearly varying pressure has increased the allowable stress by 40 to 42 percent in the 2700° to 3000° R (1500 to 1667 K) temperature range.

Figure 7 also shows how the use of a stronger clad, such as tungsten - 25-percent rhenium (W-25Re) increases the allowable stress. The Larson-Miller parameters for this material (ref. 18) are

$$B = 5.12\times10^{9}\ \text{psi} \qquad (3.52\times10^{9}\ \text{N/cm}^2)$$

$$m = -1.03\times10^{-4}/^{\circ}\text{R} \qquad (-1.85\times10^{-4}/\text{K})$$

$$J = 15$$

Although the use of any stronger clad is obviously desirable, the tube fabrication and welding of the tungsten alloys is, at present, much more difficult than for the tantalum alloys. In the present analysis most of the calculations were performed on T-222. It is possible, however, to estimate the effect of using W-25Re by observing (fig. 7) that the allowable stress for the stronger clad (i.e., W-25Re) at 3000° R (1667 K) is essentially the same as for T-222 at 2700° R (1500 K).

The realtions between the tangential or hoop stress σ and the internal pressure P can be obtained (ref. 19) from either the thick-walled relation

$$\frac{\sigma}{P} = \left(\frac{b^2 + a^2}{b^2 - a^2}\right) \tag{3}$$

or the thin-walled relation

$$\frac{\sigma}{P} = \frac{r_i}{t} \tag{4a}$$

where

r_i inner radius of tubes

t clad thickness

The thin-wall equation (eq. (4a)) can also be written in terms of the fuel-pin dimensions (fig. 6) as

$$\frac{\sigma}{P} = \frac{a}{b - a} \tag{4b}$$

When the tube diameter-to-wall thickness ratio is less than 10 (i.e., $d/t < 10$) the thick-wall equation (eq. (3)) is usually used (ref. 19). However, even for values of d/t greater than 10 the thick-wall equation (eq. (3)) is somewhat more conservative than the thin-wall equation (eq. (4b)). For example, at a b/a value of 1.10 ($d/t = 22$), the tangential stress predicted by the thick-wall equation (eq. (3)) is 5 percent larger than that calculated using the thin-wall equation (eq. (4b)); at a b/a value of 1.05 ($d/t = 42$), the stress is 2.5 percent larger. Since the thick-wall equation is more conservative than the thin-wall expression, equation (3) was used throughout the present analysis to determine the tangential clad stress and to establish the maximum allowable (end-of-life) internal pressure. That is, equations (1) and (3) are combined to obtain the maximum allowable pressure

$$P_{max} = B\left[\frac{10^J \tau_{max}}{1 - \frac{1}{mT}}\right]^{mT} \left[\frac{b^2 - a^2}{b^2 + a^2}\right] \tag{5}$$

Because of a lack of reliable information, no allowance is made for fuel swelling which may also contribute to the stresses in the clad. With the relatively large void fraction in the fuel-element designs considered, fuel swelling may be accommodated. However, more experimental data are required on this subject.

Temperature Calculations

Since the internal pressure is also influenced by the internal temperature of the fuel element, it is necessary to calculate the temperature distribution in the fuel rod. Only the rod at the maximum radial power location was considered in the analysis because the temperature and fission rate would be the greatest in that fuel rod. This combination would result in the maximum internal pressure and clad stress for a fixed fuel pin geometry and fuel loading.

Temperatures within the reactor were computed in much the same manner as in reference 1. The difference between the maximum fuel temperature T_{max} and the bulk coolant temperature T_b was calculated by the relation

$$(T_{max} - T_b)_{r,z} = \frac{F_{r,z}\varphi b}{K_1}\left[\frac{1}{2} - \frac{\ln\left(\frac{a}{c}\right)}{\left(\frac{a}{c}\right)^2 - 1} + \frac{K_1}{K_2}\ln\left(\frac{b}{a}\right) + \frac{K_1}{hb}\right] \tag{6}$$

where

$F_{r,z}$	local to average power factor at r and z
φ	average heat flux
a, b, c	fuel rod dimensions (fig. 6)
K_1	thermal conductivity of fuel
K_2	thermal conductivity of clad
r	radial location
z	axial location
h	heat transfer coefficient

The heat-transfer coefficient h was determined from the relations for parallel flow of liquid-metal through rod bundles (ref. 20). No attempt was made to account for circumferential variations in the clad temperature (ref. 21) which might result from nonuniform circumferential heat transfer associated with closely spaced rods. Additional studies must be performed to determine the magnitude of the temperature variations and to establish their effect on stress and strain in the clad.

In the present study, exit coolant temperatures in the range 2400° to 3000° R (1333 to 1667 K) were considered. The coolant-temperature rise across the reactor core was assumed to be 100° R (55.5 K). The reactor power was arbitrarily fixed at 3 megawatts (thermal); however, the required size for a reactor with the same specific power but with another power level can be approximated by replotting the curves using the following proportionality constant

$$Y_2 = Y_1 \left(\frac{P_2}{P_1}\right)^{1/3} \tag{7}$$

where

Y_1 reactor diameter and length (L/D = 1) as plotted herein

P_1 power level used in this study (i.e., 3 MW_t)

P_2 desired power level

Y_2 required diameter and length of reactor at desired power level

As an example, suppose an increase in power to 6 megawatts were required for a particular application being studied. Then a point, which is plotted in this report at a diameter and length of 10 inches (25.4 cm), would be shifted by the cube root of 2.0 (i.e., 6 MW_t/3 MW_t) or to a diameter and length of 12.6 inches (32 cm). A similar shift of all of the points along a given curve for the present 3-megawatt study would establish the corresponding curve for the required 6-megawatt power level. The nuclear criticality curves are, of course, independent of power level and would, therefore, not be changed. The actual operating point would be established by the intersection of the criticality curve with the 6-megawatt curve.

VENTED FUEL ELEMENT DESIGN

The vented fuel concept (fig. 1) has the advantage of being the only one considered in which the basic fuel-element design is independent of the fission-gas release rate and the total operating time. This is because the gases are vented from the fuel element and there is essentially little or no internal pressure buildup resulting from accumulation of gaseous-fission products as long as the vent tube remains open and the external storage volume is properly designed and adequately cooled. Since no appreciable internal pressure will result from gas buildup in the vented concept, the stresses in the clad are independent of the percentage of fission gas released and the operating time of the reactor. Therefore, the minimum clad thickness, commensurate with manufacturing feasibility and corrosion effects, can be used in this concept. For the cases considered in this study, it was assumed that a minimum clad thickness of 0.020 inch (0.508 mm) would be required to resist long-term corrosion effects of the liquid-metal coolant.

In the absence of a limiting clad stress in the vented fuel-pin design, heat transfer and nuclear criticality become the criteria with which the required core size can be established. The nuclear criticality requirements for this general type of reactor have been established (ref. 1) over a reasonable range of fuel, clad, and coolant compositions. Heat-transfer variables that could prove to be limiting conditions in this type of design are the allowable heat flux and the allowable fuel temperature. For the power densities of interest in the present study, the maximum heat flux is less than 5×10^5 Btu per hour per square foot (158 W/cm^2) which is less than the critical heat flux for most liquid-metal coolants (ref. 1). The allowable fuel temperature, therefore, is the variable which was used to establish the core size of the vented fuel concept.

Uranium dioxide has a reported melting point (ref. 10) of $5489\pm54^\circ$ R (3033 ± 30 K) although melting points as low as 4824° R (2680 K) and as high as 5675° R (3153 K) have been experimentally observed (ref. 10). These wide variations in observed melting points are attributed to a combination of small quantities of impurities, the high-vapor pressure of UO_2, and the difficulties in measuring and controlling high-temperature experiments (ref. 10). Regardless of the cause of these variations, it would appear that a maximum fuel temperature of 4800° R (2667 K) would be a reasonable upper limit if melting of the fuel were to be avoided. Of course, other factors may influence the final choice of an allowable fuel temperature for the vented fuel pin. Possible plugging of the vent tube and fuel losses from the core, for example, might limit the operating temperature to a value less than 4800° R (2667 K).

The maximum fuel temperature actually achieved in the reactor depends on power density and the fuel thickness. For a given set of operating parameters (e.g., coolant flow rate, power level, power distribution, coolant temperature), fuel temperature is, therefore, a function of both the pin size and core size. Figure 8 shows how the core size varies as a function of fuel loading for pin sizes ranging from 0.25 to 1.0 inch (6.35 to 25.4 mm). In these calculations, a maximum allowable fuel temperature of 4800° R (2667 K) was used; the exit coolant temperature was 2700° R (1500 K); the coolant flow rate was chosen to give a 100° R (55.5 K) temperature rise across the core and the coolant velocity and core pressure drop were allowed to vary to satisfy this constraint. The fuel loading was varied over the permissable range of thicknesses for each pin size; that is, from a trivial case of zero fuel thickness to the maximum thickness which could physically be accommodated by a particular pin size. (A flow chart of the calculational procedure used is given in the appendix.)

The results of these calculations (fig. 8) show that the core size necessary to satisfy both the fuel-temperature limitation and the nuclear criticality limit of $U^{233}O_2$, varies

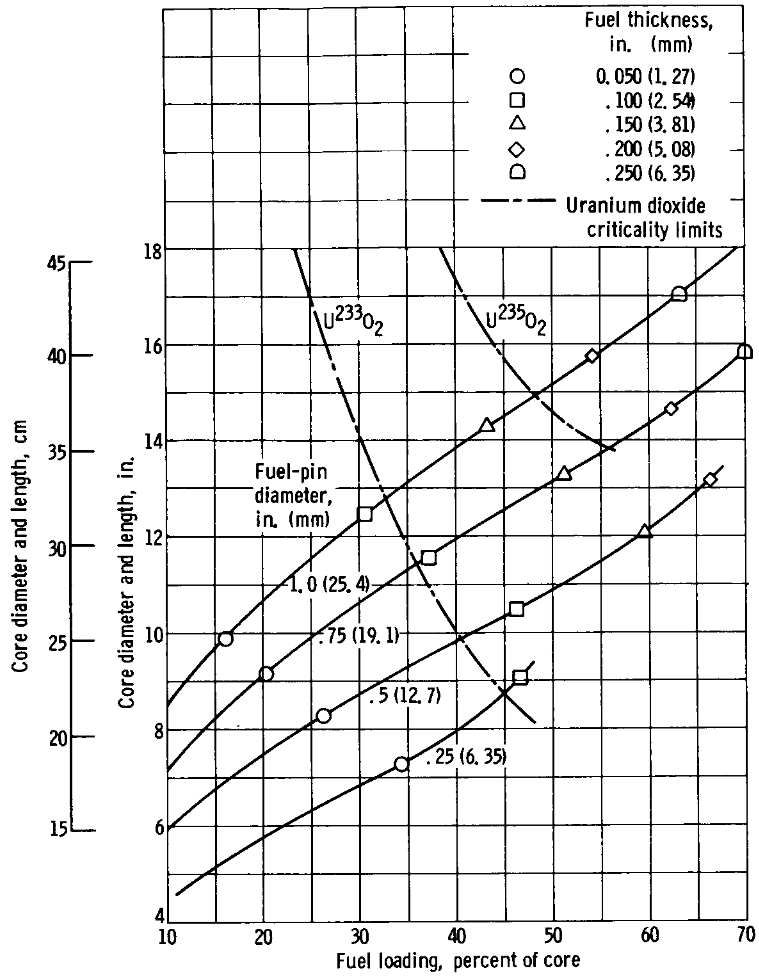


Figure 8. - Effect of uranium dioxide fuel loading and pin diameter on required core size of vented fuel concept. Core power, 3 megawatts; maximum fuel temperature, 4800° R (2667 K); coolant temperature, 2700° R (1500 K); equivalent diameter, 0.125 in. (3.175 mm); clad thickness, 0.020 in. (0.51 mm).

from about 9 inches (22.9 cm) for the 0.25-inch (6.35-mm) fuel pin to nearly 13 inches (33.0 cm) for a 1.0-inch (25.4-mm) fuel pin. For $U^{235}O_2$ fuel, the required core size varies from 14 to 15 inches (35.6 to 38.1 cm) for fuel pins with diameters of 0.75 to 1.0 inch (19.1 to 25.4 mm).

Superimposed on the curves shown in figure 8 are particular fuel thicknesses required to obtain the fuel loadings for the various pin sizes. It is of interest to note that, when the nuclear criticality limits are satisfied, the required fuel thickness is nearly the same for all pin sizes (e.g., the required fuel thickness is about 0.100 in. (2.54 mm) for $U^{233}O_2$ fuel). This is an important factor to consider in the final selection of the proper pin size to be used for the vented concept. Figure 9 shows how the fuel thickness

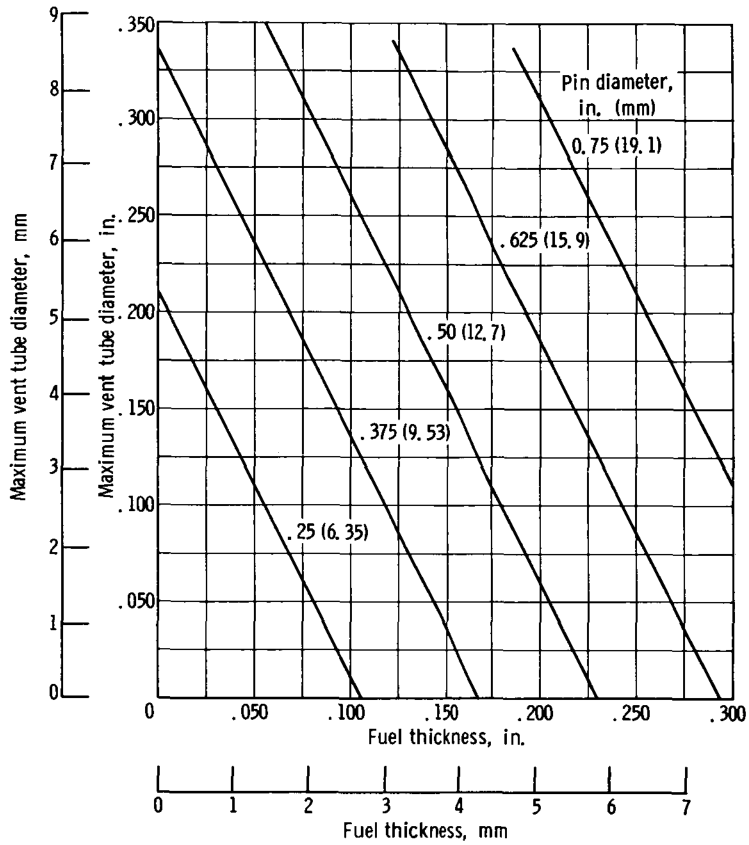


Figure 9. - Effect of uranium dioxide fuel thickness and pin diameter on allowable vent tube diameter for clad thickness of 0.02 inch (0.51 mm).

influences the allowable vent-tube diameter for various fuel-pin sizes. The limiting values as demonstrated by figure 9 are the case where the sum of the clad and fuel is equal to fuel pin radius, and the case where the fuel thickness is zero. In the first case, there is no central void volume available into which the vent tube can be inserted (fig. 1); in the second case, the vent tube diameter can be as large as the inside diameter of the clad. For a practical design, the pin diameter and fuel thickness should be selected so that the vent tube is a reasonable size. For example, if the 0.25-inch (6.35-mm) diameter fuel pin (fig. 8) with a fuel thickness of 0.100 inch (2.54 mm) were selected, the maximum allowable vent tube (fig. 9) that could be used with this combination of pin diameter, clad thickness, and fuel thickness is 0.01 inch (0.25 mm). Because this size vent tube would be difficult to fabricate and would be more apt to plug than a larger diameter tube, the use of this particular geometry is impractical. By going to a larger diameter fuel pin (and a larger core size), the allowable vent-tube diameter can be increased to a more useful size; for example, for a 0.50-inch (12.7-mm) pin, the maximum allowable vent tube diameter (fig. 9) would be over 0.25 inch (6.35 mm) for the same fuel thickness.

Another factor, which also influences the selection of the proper core size, is fuel burnup. As noted previously, the vented fuel-element design is not directly dependent on fission-gas release. Even with vented fuels, however, fuel swelling and/or changes in nuclear criticality due to fuel depletion could be limiting factors in determining the maximum power density that can be achieved for long-life cores. Again using the 0.25-inch (6.35-mm) diameter fuel-pin case from figure 8 as an example, a 9-inch (22.9-cm), 3-megawatt core with a 45-volume-percent UO_2 fuel loading has a 15-percent burnup for 50 000 hours of operation (fig. 4). If the core size was increased to 11 inches (27.9 cm) or more using a 0.75-inch (19.1-mm) diameter fuel pin (fig. 8), burnup could be reduced to less than 10 percent for the same operating conditions.

The allowable fuel temperature also has a strong influence on the choice of a fuel-pin diameter for the vented fuel concept. Figure 10 shows the required core sizes for sever-

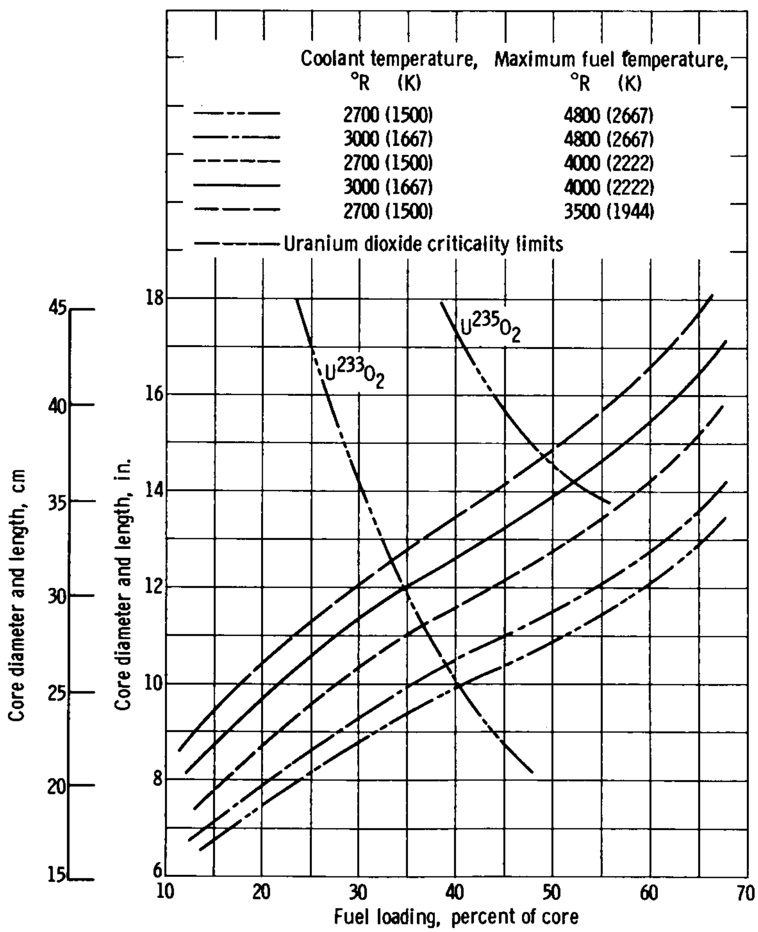


Figure 10. - Effect of uranium dioxide fuel loading and operating temperature on required core size of the vented fuel concept. Core power, 3 megawatts; pin diameter, 0.5 inch (12.7 mm); equivalent diameter, 0.125 inch (3.175 mm); clad thickness, 0.020 inch (0.51 mm).

eral combinations of coolant temperatures and allowable fuel temperatures using a 0.50-inch (12.7-mm) fuel pin. As the difference between the allowable fuel temperature and the coolant temperature decreases, the power density of the core must also decrease. This decrease in power density causes a corresponding increase in core size. For the conditions shown in figure 10, the required core sizes for a 0.50-inch (12.7-mm) fuel pin varies from 10 to nearly 13 inches (25.4 to 33.0 cm) using $U^{233}O_2$ fuel. Similar variations in required core size would occur for other fuel pin sizes.

UNVENTED FUEL ELEMENT DESIGN

From a fabrication and operating viewpoint, the unvented fuel element (fig. 2) is probably the least complicated of the three concepts considered in this study. To obtain the desired behavior in this design, it is only necessary to provide sufficient void space into which the fission gases can collect. However, because all the fission products of an individual pin are contained within that fuel pin, the internal pressure could be quite large and the cladding must be capable of withstanding the imposed stresses. To minimize these stresses and the resulting creep which will occur, it is necessary to either thicken the clad or reduce the internal pressure.

For a given clad thickness and a given quantity of gaseous fission products, the pressure stresses can be decreased by increasing the available void space. Actually, any attempt to reduce the pressure stresses in this manner helps in two ways: (1) it provides a greater void volume for the fission gas, and (2) it decreases the temperature of the gas. Condition (2) is a direct result of condition (1) because an increase in void volume can only be accomplished by either reducing the fuel thickness or by reducing the power density (i.e., increasing the size of the reactor).

Characteristics of Unvented Fuel Pin

For any size fuel pin, operating under a fixed set of conditions, there are several possible combinations of clad, fuel, and void which would satisfy the 1-percent creep requirement. Figure 11 illustrates this point. The relative composition of the three components (clad, fuel, and void) are shown for a 0.5-inch (12.7-mm) diameter pin with an arbitrary set of operating conditions. As the clad thickness is increased, the required void obviously decreases because the thicker clad is stronger and can withstand higher internal pressure. After the clad and void requirements have been satisfied, the remaining volume can then be used for the fuel (UO_2).

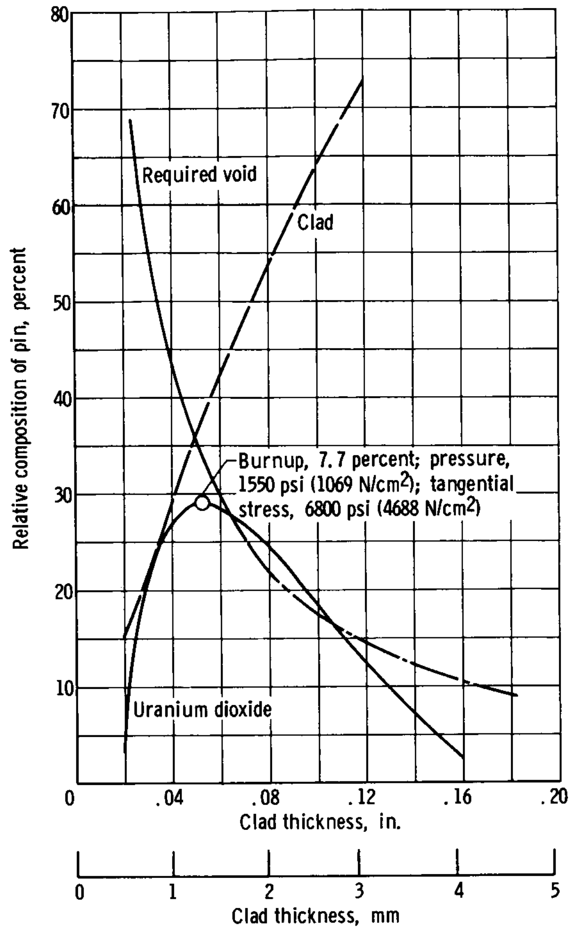


Figure 11. - Typical variation in pin composition with clad thickness for 1-percent creep in T-222 cladding. Reactor lifetime, 50 000 hours; bulk temperature, 2700° R (1500 K); heat flux, 150 000 Btu per hour per square foot (47 W/cm^2); pin diameter, 0.50 inch (12.7 mm); equivalent diameter, 0.125 inch (3.175 mm).

The limiting internal pressure that the cladding can ultimately withstand can be obtained from equation (3). As the clad approaches its maximum thickness (i.e., as a approaches zero in eq. (3)), the ratio of the tangential stress to the internal pressure is found to be 1.0. The maximum or limiting internal pressure is therefore equal to the maximum allowable stress for that type of cladding at the operating temperature being considered. In practice, since the quantity of fission gas per fuel pin is fixed for a given power density, the internal pressure would become quite large as the thickness of the clad approached the maximum value. Therefore, the limiting pressure which can be accommodated in the unvented fuel pin will be less than that which is theoretically feasible (i.e., $P_{max} < \sigma_{max}$). In the analysis of the unvented concept (see appendix for calculational procedure), the cladding thickness was allowed to vary over the entire range of possible solutions. The maximum clad thickness considered is the point where no additional fuel can be removed to provide additional void volume (i.e., at zero fuel volume).

The optimum pin composition obtained from these calculations is the one that maximizes the fuel loading, since this should result in the smallest reactor. In the example selected (fig. 11), the optimum clad thickness was found to be 0.052 inch (1.32 mm) and the relative clad, fuel, and void composition were 37, 29, and 34 percent, respectively. The maximum fuel burnup was 7.7 percent, and the internal pressure and clad stress were 1550 and 6800 psi (1069 and 4688 N/cm^2), respectively, for the operating conditions chosen.

Figures 12 and 13 show how the temperature, pressure, and stress within the fuel pin change with variation in the clad thickness. A short discussion on the behavior of each of these variables may be worthwhile.

Clad temperature. - Since the coolant temperature, coolant flow rate, and the heat flux are assumed to be constant, the average clad temperature depends on the thickness of the clad. Because of the high thermal conductivity of the clad material, only a rela-

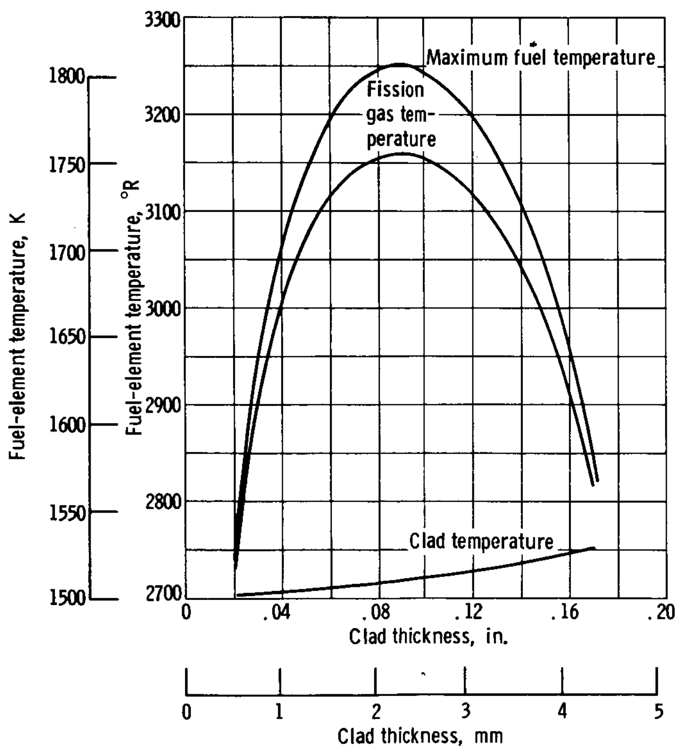


Figure 12. - Variation in uranium dioxide fuel-element temperature with clad thickness. Creep limited to 1 percent in T-222 cladding. Reactor lifetime, 50 000 hours; bulk temperature, 2700° R (1500 K); heat flux, 150 000 Btu per hour per square foot (47 W/cm^2); pin diameter, 0.50 inch (12.7 mm); equivalent diameter, 0.125 inch (3.175 mm).

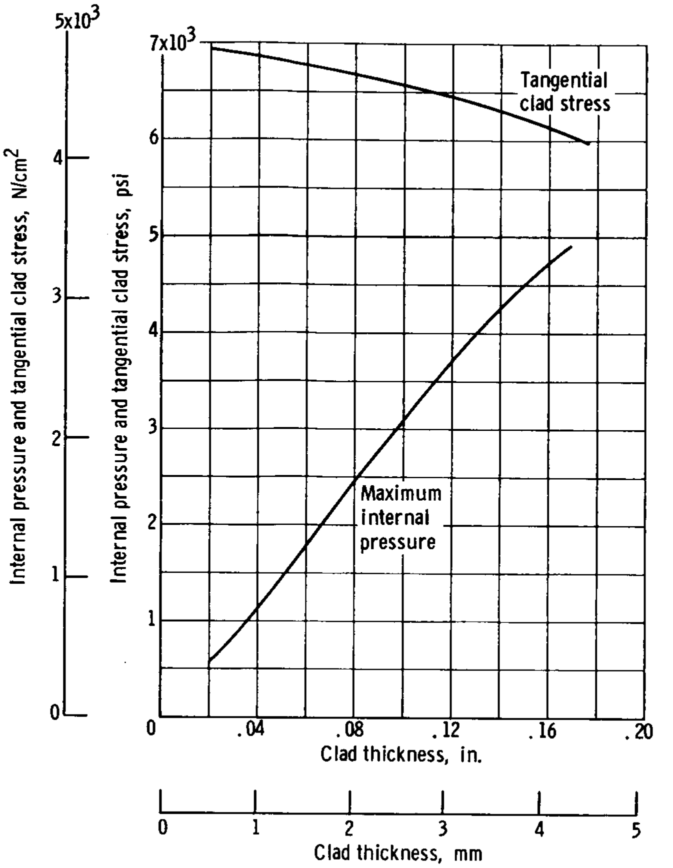


Figure 13. - Variation in tangential clad stress and internal pressure with clad thickness for conditions at end of life. Creep limited to 1 percent in T-222 cladding. Reactor lifetime, 50 000 hours; bulk temperature, 2700° R (1500 K); heat flux, 150 000 Btu per hour per square foot (47 W/cm^2); pin diameter, 0.50 inch (12.7 mm); equivalent diameter, 0.125 inch (3.17 mm); fuel, uranium dioxide.

tively small increase in clad temperature occurs as the clad thickness is increased from 0.02 inch (0.5 mm) to over 0.16 inch (4 mm).

Maximum fuel temperature. - As the clad thickness is increased, the fuel volume first increases and then decreases as the clad volume becomes larger. The maximum fuel temperature follows the same general trend except that the peak temperature occurs at the point where the combined thermal resistance of clad and fuel is greatest.

It should be noted that the maximum fuel temperature is considerably lower than 4800° R (2667 K) because of the relation between fuel temperature, internal gas pressure, and clad stress. At the optimum composition in the example chosen herein (figs. 11 to 13), the maximum fuel temperature is 3150° R (1750 K); any attempt to increase this fuel temperature by either increasing the power density or fuel thickness, will result in an increase in the gas pressure. This increase in the internal pressure will, in turn, cause the stress in the cladding to increase beyond its allowable value.

Fission-gas temperature. - The gas temperature (fig. 12) follows the parabolic behavior of the maximum fuel temperature but is slightly lower. This is due to the nonuniform axial temperature distribution, which is common to most nuclear-reactor fuel elements. Since the pressure of the fission gas contained within the central cavity of the fuel element is more dependent on the average internal fuel-element temperature than on a local (maximum) fuel temperature, the calculated internal fuel temperatures were integrated along the length of the fuel element to obtain the average fission-gas temperature shown on figure 12.

One assumption made in this particular calculation (which could prove to be unconservative) was that the diameter of the central void was uniform along the entire length of the fuel element. It is possible that the cavity will, in fact, be irregular. For example, the cavity could be elliptical with the fuel being thicker nearer the ends of the fuel pins and thinner in the middle. With this configuration, the temperature of the fuel surrounding the cavity might be nearly constant, in which case the average gas temperature would be essentially the same as the maximum fuel temperature.

Clad stress. - The allowable tangential clad stress necessary to limit the creep to 1 percent decreases with increasing clad thickness (see fig. 13) because of the higher clad temperature which results with the thicker clad.

Internal pressure. - The maximum internal pressure which the clad can tolerate without exceeding the stress limit obviously increases with clad thickness (fig. 13). In the limit, the internal pressure approaches the allowable clad stress. In practice, the internal pressure can be increased to the point where no additional fuel can be removed (i.e., zero fuel volume) to provide additional void space.

Optimum Fuel Loading and Required Reactor Size

The behavior of the variables discussed in the previous example are typical of the unvented fuel element. Figure 14 shows, for example, the variation in pin composition for a completely different set of conditions. Although the absolute values differ, the characteristic shapes of the curves are quite similar to those of the previous example (fig. 11). However, there is a significant increase in the optimum fuel loading (i.e., to 54 percent of the fuel pin). This increase in the available fuel volume is due chiefly to the reduced operating time (20 000 hr) used in this second example (fig. 14).

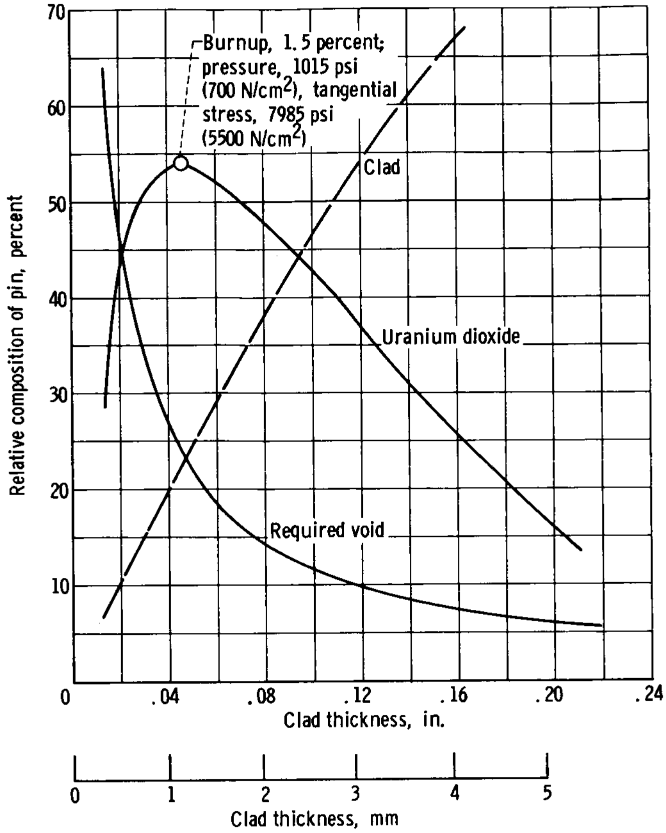


Figure 14. - Typical variation in pin composition with clad thickness for 1-percent creep in T-222 cladding. Reactor lifetime, 20 000 hours; bulk temperature, 2700° R (1500 K); heat flux, 2×10^{5} Btu per hour per square foot (63 W/cm^2); pin diameter, 0.75 inch (19.1 mm); equivalent diameter, 0.125 inch (3.175 mm).

The way in which the operating temperature and time alters the optimum fuel concentration can be seen on figure 15. Over the range of conditions shown, the optimum fuel composition varies from 21 to 68 percent of the fuel-pin volume, while the optimum clad thickness drops from 0.09 to 0.03 inch (2.29 to 0.76 mm) over the same range. It is of interest to note that, for the particular clad material used in this example (i.e., T-222), a change in operating temperature of 300° R (167 K) is approximately equivalent to the difference between operating for 20 000 or 50 000 hours. This 300° R (167 K) temperature difference also represents the approximate difference between using T-222 and W-25 Re (fig. 7). A 50 000-hour, 3000° R (2667 K) core using W-25 Re clad would, therefore, have about the same characteristics as the 50 000-hour, 2700° R (1500 K) curve for T-222 (fig. 15).

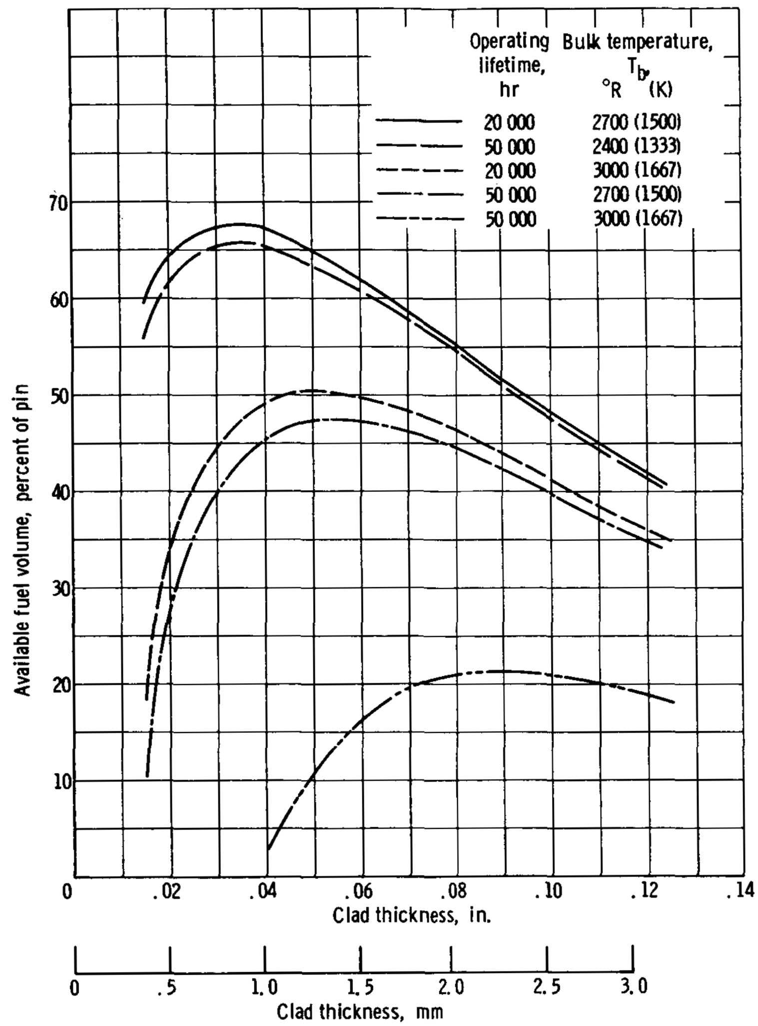


Figure 15. - Effect of clad thickness, clad temperature, and core lifetime on available fuel volume for 1-percent creep in T-222 cladding. Heat flux, 100 000 Btu per hour per square foot (31.5 W/cm^2); pin diameter, 0.75 inch (19.1 m); fuel, uranium dioxide.

Figure 16 shows how the optimum fuel composition changes with heat flux. As the heat flux is increased, the gas temperature and gas pressure also increase. To maintain the required stress level, the clad thickness must also be increased. The combination of these factors results in a reduction in fuel volume as the heat flux is increased to higher and higher values.

If a particular reactor power level is desired, the required reactor size can easily be determined from the power level, heat flux, and fuel-pin geometry. For example, for the 2.0×10^5 Btu per hour per square foot (63 W/cm^2) heat flux case on figure 16, a 3-megawatt reactor would have to be 15.2 inches (38.6 cm) in diameter and length (L/D = 1) to satisfy all of the geometric constraints used in this example. By determining

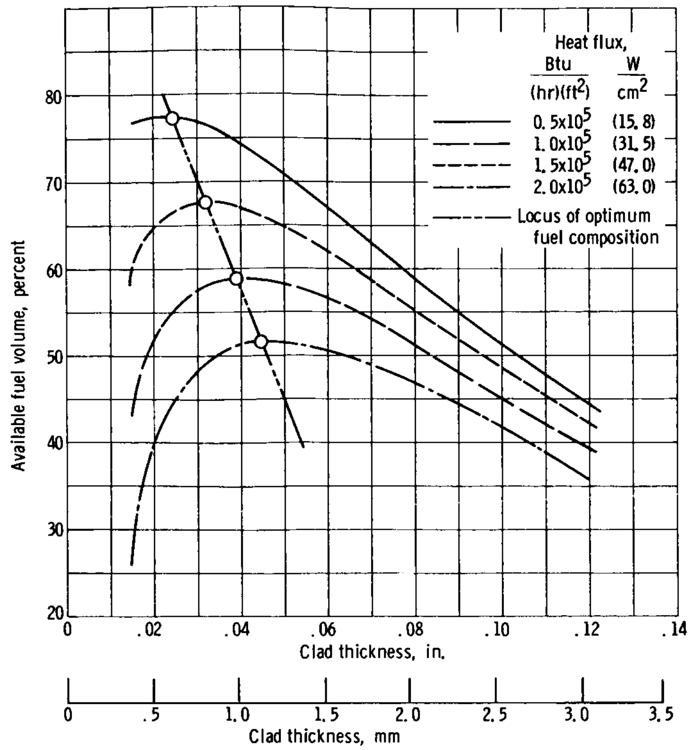


Figure 16. - Effect of clad thickness and heat flux on available fuel volume for 1-percent creep in T-222 cladding. Reactor lifetime, 20 000 hours; bulk temperature, 2700° R (1500 K); pin diameter, 0.75 inch (19.1 mm); equivalent diameter, 0.125 inch (3.175 mm); fuel, uranium dioxide.

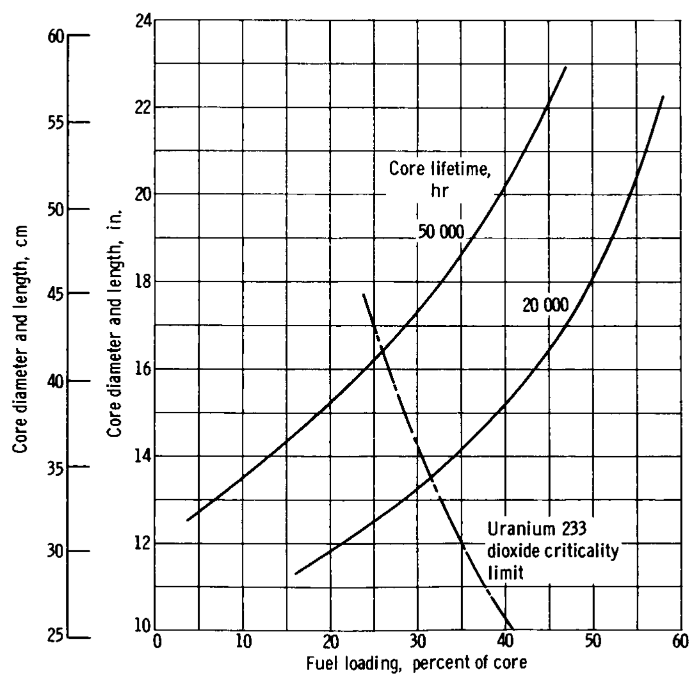


Figure 17. - Effect of operating time on required core size. Creep limited to 1 percent in T-222 cladding. Thermal power output, 3 megawatts; spacing-to-diameter ratio, 1.08; bulk temperature, 2700° R (1500 K); pin diameter, 0.75 inch (19.1 mm).

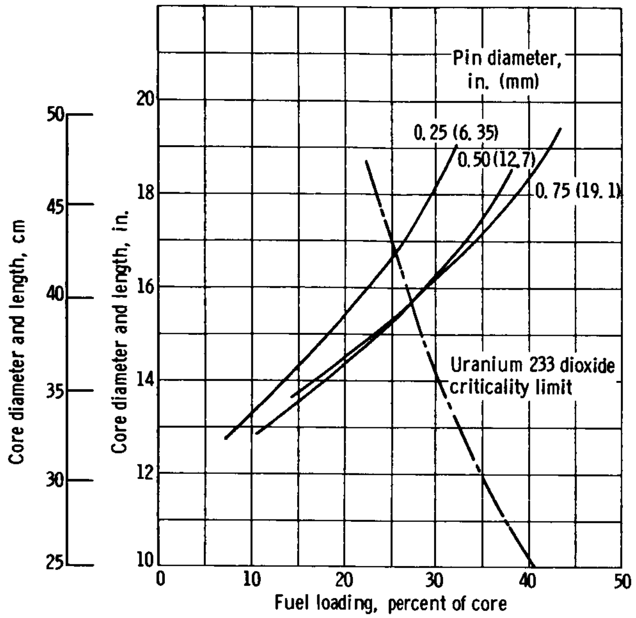


Figure 18. - Effect of pin diameter on required core size for constant equivalent diameter and 1-percent creep in T-222 cladding. Reactor lifetime, 50 000 hours; bulk temperature, 2700° R (1500 K); equivalent diameter, 0.125 inch (3.175 mm), thermal power output, 3 megawatts.

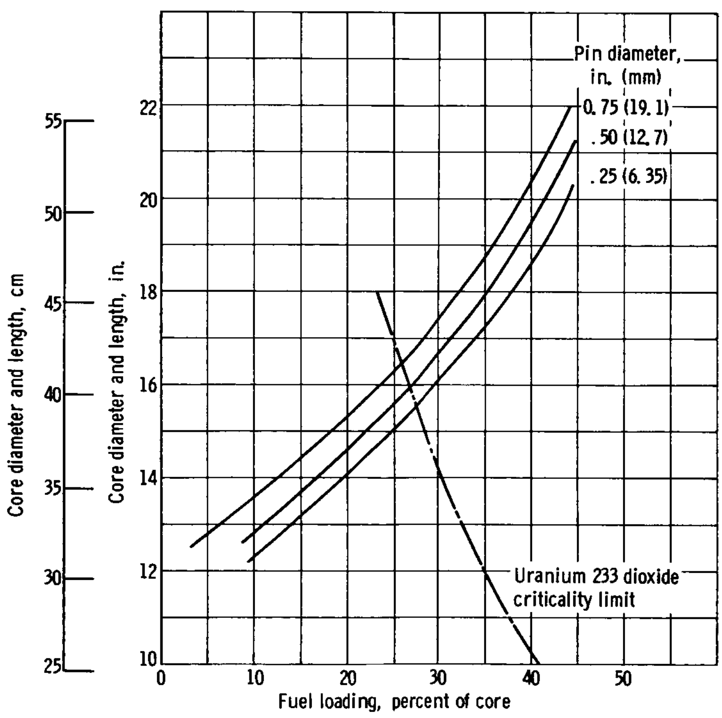


Figure 19. - Effect of pin diameter on required core size at constant coolant fraction and 1-percent creep in T-222 cladding. Thermal power output, 3 megawatts; reactor lifetime, 50 000 hours; bulk temperature, 2700° R (1500 K); coolant fraction, 0.222; spacing-to-diameter ratio, 1.08.

the core size and optimum composition for a variety of cases, it is possible to determine the minimum reactor size of an unvented fuel-pin design which satisfies both nuclear and stress (creep) limitations. Figure 17 is a plot of required core size against optimum fuel loading for a 3-megawatt reactor at operating lifetimes of 20 000 and 50 000 hours. The intersection of the criticality curve ($U^{233}O_2$) and the creep curve determine the minimum core size which can be achieved. For the 0.75-inch (1.91-cm) diameter fuel pin used in this example, there is about a 3-inch (7.6-cm) difference in the required core size between operation for 20 000 or 50 000 hours.

Effect of fuel pin-diameter. - Figures 18 and 19 show the effect of fuel-pin diameter on the required core size of a 50 000-hour unvented fuel pin. Figure 18 shows the case in which the pin diameter was varied from 0.25 to 0.75 inch (6.35 to 19.1 mm), while the equivalent hydraulic diameter was held constant at 0.125 inch (3.175 mm). Although the core size for the smallest pin is slightly larger, there is essentially no difference in the resulting core size of the two larger diameter pins. The required reactor size is about 15.7 inches (39.9 cm) for both the 0.50- and 0.75-inch (12.7- and 19.1-mm) fuel pins.

Rather than holding the equivalent diameter constant, a more realistic approach would be to hold the coolant fraction constant (i.e., a constant spacing-to-pin diameter ratio) and allow the equivalent diameter to vary proportionally with pin diameter. Figure 19 shows that, under these conditions, there is an advantage in using small diameter pins. The required core sizes using 0.25-, 0.50-, and 0.75-inch (6.35-, 12.7-, and 19.1-mm) diameter pins are approximately 15.5, 16, and 16.5 inches (39.4, 40.6, and 41.9 cm), respectively. The spacing-to-diameter ratio was held at 1.08 for these calculations (fig. 19).

Although the use of the small 0.25-inch (6.35-mm) diameter rods results in a smaller reactor core (fig. 19) when the spacing-to-diameter ratio (or coolant fraction) is held constant, there are several other factors that should be considered in selecting the fuel-pin diameter. Figure 20 is a plot of the available fuel volume for 0.25-, 0.50-, and 0.75-inch (6.35-, 12.7-, and 19-mm) diameter pins for an assumed set of conditions. The figure shows that, for the same size reactor, the available fuel volume is inversely proportional to the pin diameter because of the better strength characteristics of small pins. However, the optimum clad thickness for the 0.25-inch (6.35-mm) rods is only 0.012 inch (3 mm). This is quite thin considering possible fabrication and long-term corrosion problems. The width of the parabolic peak about the optimized clad thickness is also very narrow for the small fuel pins, which indicates a degree of sensitivity to possible manufacturing variations in the clad thickness.

Another area where manufacturing tolerances might be critical for smaller fuel pins is the close spacing between pins. Because the spacing-to-diameter ratio is held constant ($s/d = 1.065$) in this comparison (fig. 20), the clearance between the 0.25-inch

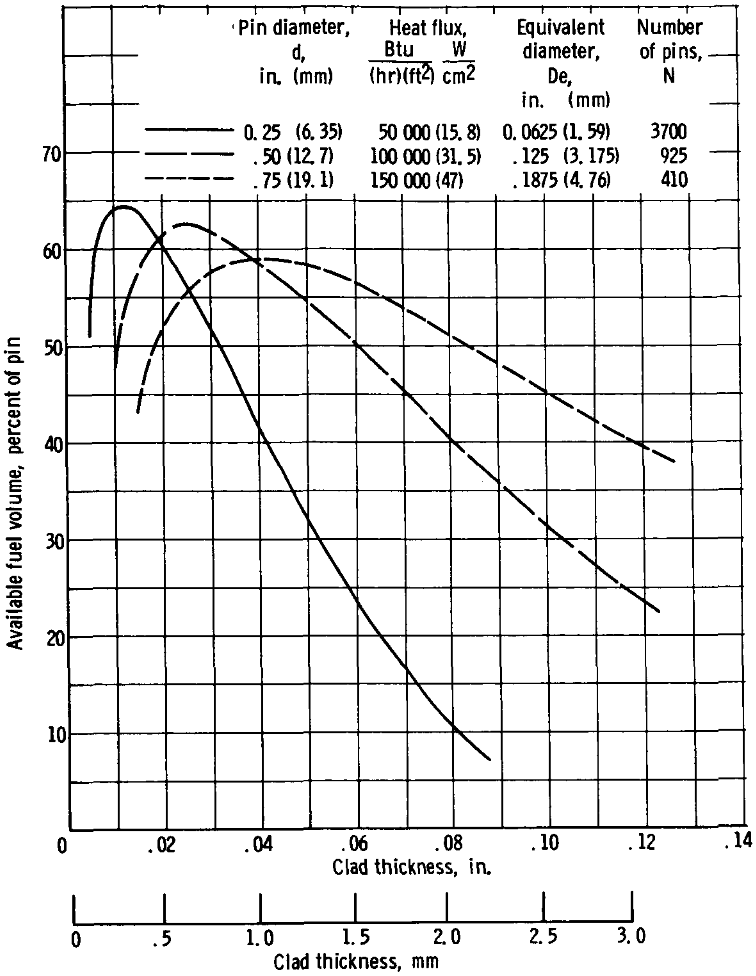


Figure 20. - Effect of clad thickness and pin diameter on available fuel volume at constant core size. Creep limited to 1 percent in T-222 cladding. Cooling volume, 20 percent of core; core diameter, 17 inches (43.2 cm); reactor lifetime, 20 000 hours; bulk temperature, 2700° R (1500 K); spacing-to-diameter ratio, 1.065; fuel, uranium dioxide.

(6.35-mm)diameter pins is only 0.016 inch (4 mm). Considering the number of pins in this design (3700), the assembly of the pins into a complete reactor core would be difficult if only a small dimensional tolerance were allowed.

In view of these factors, it might be better, therefore, to select a larger diameter pin even though the resulting core size would be somewhat larger than with the smaller pin.

Effect of pin spacing. - In addition to the assembly problems caused by closely spaced pins, there is the circumferential heat-transfer problem (ref. 21) which is associated with the flow of liquid-metal in a closely packed array of rods. This circumferential variation in heat transfer causes local temperature variations in the clad (ref. 22) which, in turn, could produce thermal stresses of considerable magnitude (ref. 23). The magnitude of the circumferential temperature variations depends on several factors such as the thermal conductivity of the clad and fuel material. There are no experimental data presently available on these circumferential effects. However, analytical predictions (refs. 21 to 25) indicate that the spacing-to-diameter ratio should be about 1.10 or greater in order to avoid very large variations. At spacing-to-diameter ratios greater than 1.35, for example, circumferential variations are negligible (refs. 20 and 22).

Although it appears that is would be much better to have large spacing-to-diameter ratios in a nuclear reactor, there are other considerations that contradict this premise. To obtain a small, compact reactor, for example, nuclear criticality dictates that the fuel pins be closely spaced in order to maximize fuel content. (At a spacing-to-

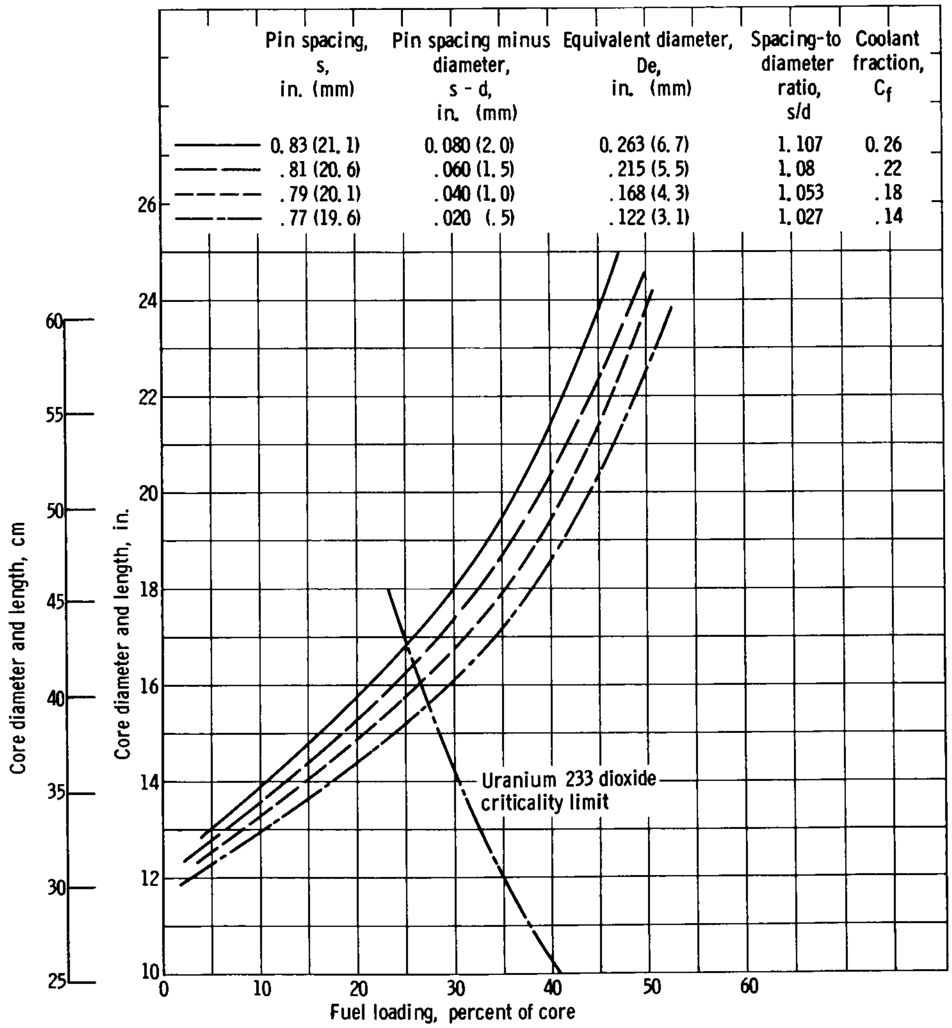


Figure 21. - Effect of pin spacing on required core size for 1-percent creep in T-222 cladding. Pin diameter, 0.75 inch (19.1 mm); reactor lifetime, 50 000 hours; bulk temperature, 2700° R (1500 K); thermal power output, 3 megawatts.

diameter ratio of 1.35 the coolant fraction for pins in a triangular array is over 50 percent of the core volume; at a spacing-to-diameter ratio of 1.10, the coolant fraction is 25 percent.)

Another consideration in a fast reactor is that any gross movement between fuel pins could result in unstable and possibly unsafe reactor operation. The solution to prevent such an occurrence might be to tightly band the core, forcing the fuel pins to remain in a fixed relation to each other. This would, of course, also indicate a need for closely spaced fuel pins and shows that, although large spacing-to-diameters may be advantageous for certain reasons, the final reactor design will have to compromise in this area.

In this study no calculations of circumferential variations in heat transfer or in the resulting thermal stresses were made. Additional work must obviously be performed because such stresses must be superimposed on the pressure stresses already considered and the resulting combined stress could result in severe local distortions of the clad at the temperatures and operating times of interest.

Neglecting circumferential variations, figure 21 shows how the spacing between fuel pins affects the required core size for the unvented fuel-pin design. Over a range of spacing-to-diameter ratios (1.027 to 1.107) the required core size for a 0.75-inch (19.1-mm) diameter fuel pin varies from 15.5 to 17 inches (39.4 to 43.2 cm). This is approximately a 1/2-inch (1.27-cm) increase in core size for each additional 0.02-inch (0.5-mm) change in pin spacing. Similar trends were found for other size fuel pins. Table I shows, for example, representative conditions for 0.25-, 0.50-, and 0.75-inch (6.35-, 12.7-, and 19.1-mm) diameter fuel pins at a spacing-to-diameter ratio of 1.08. It is felt that this spacing (i.e., s/d = 1.08) is about as low as can be tolerated in view of the uncertainties in the circumferential heat-transfer behavior.

<u>Effect of fuel material</u>. - Most of the previous calculations were performed assuming that the fuel was $U^{233}O_2$. However, it is possible that other ceramic fuel materials could be used in a reactor of this type. Uranium nitride (UN), for example, appears to offer some advantages over UO_2. Uranium nitride has a higher fuel density, and the thermal conductivity is an order of magnitude greater than UO_2 (ref. 17). But, there is a chemical reaction between uranium nitride and tantalum (ref. 17) at temperatures in excess of 2300° R (1278 K). The use of a barrier or a cladding other than T-222 would, therefore, be required in order to utilize UN as a fuel.

Although the use of uranium 233 (U^{233}), instead of uranium 235 (U^{235}) decreases the size of the reactor significantly (ref. 1), the ability to manufacture large quantities of uranium 233 fuels is presently hampered by the high-intensity gamma radiation of these fuels. Special equipment and remote handling techniques are being developed (ref. 26) to overcome this difficulty, but it is possible that in the immediate future the fabrication of uranium 233 fuel elements may be difficult and expensive.

TABLE I. - EFFECT OF FUEL-PIN DIAMETER ON SOME TYPICAL PARAMETERS

IN UNVENTED REACTOR USING URANIUM 233 DIOXIDE FUEL,

T-222 CLAD, AND LITHIUM 7 COOLANT

Parameters	Fuel-pin diameter, in.; cm		
	0.25; 0.64	0.50; 1.27	0.75; 1.91
Operating Conditions			
Power level, MW_t	3	3	3
Core lifetime, hr	50 000	50 000	50 000
Exit coolant temperature, ^{o}R; K	2700; 1500	2700; 1500	2700; 1500
Coolant temperature rise, ^{o}R; K	100; 55.5	100; 55.5	100; 55.5
Peak-to-average power factor	1.7	1.7	1.7
Allowable clad strain, percent	1.0	1.0	1.0
Geometry			
Pin diameter, in.; cm	0.25; 0.64	0.50; 1.27	0.75; 1.91
Spacing-to-diameter ratio	1.08	1.08	1.08
Spacing, in.; cm	0.27; 0.69	0.54; 1.37	0.81; 2.06
Interpin gap, in.; mm	0.02; 0.5	0.04; 1.0	0.06; 1.5
Equivalent hydraulic diameter, in.; cm	0.071; 0.18	0.143; 0.36	0.215; 0.55
Reactor diameter and length, in.; cm	15.4; 39.1	16.0; 40.6	16.5; 41.9
Number of fuel pins	2941	793	376
Clad thickness, in.; mm	0.025; 0.64	0.048; 1.22	0.070; 1.78
Fuel thickness, in.; mm	0.068; 1.73	0.137; 3.48	0.213; 5.41
Core Composition			
Fuel fraction	0.272	0.271	0.263
Clad fraction	0.279	0.273	0.263
Void fraction	0.227	0.234	0.252
Coolant fraction	0.222	0.222	0.222
Miscellaneous Conditions			
Maximum heat flux, Btu/(hr)(ft^2); W/cm^2	70 000; 22	125 000; 39	170 000; 54
Maximum fuel temperature, ^{o}R; K	2830; 1572	3150; 1750	3570; 1983
Maximum clad temperature, ^{o}R; K	2705; 1503	2710; 1505	2720; 1511
[a]Internal pressure, psia; N/cm^2	1530; 1060	1450; 1010	1365; 950
[a]Clad stress, psi; N/cm^2	6970; 4840	6850; 4760	6700; 4650
Maximum strain, percent	1.0	1.0	1.0
Average fuel depletion, percent	6.0	5.4	5.0
Coolant flow rate, lb/sec; kg/sec	28.5; 12.9	28.5; 12.9	28.5; 12.9
Coolant velocity, ft/sec; m/sec	3.8; 1.16	3.6; 1.10	3.3; 1.01
Frictional pressure drop, psi; N/cm^2	0.2; 0.14	0.1; 0.07	<0.1; <0.07

[a]End-of-life condition.

Figure 22 indicates how the choice of fuel material affects reactor size for a typical set of conditions. The use of $U^{233}N$ instead of $U^{233}O_2$ reduces the required reactor size by approximately 1 inch (2.54 cm). This reduction in size is the combined result of the higher thermal conductivity and better nuclear characteristics of UN.

An increase in reactor size of over 2 inches (5.08 cm) results from the use of $U^{235}O_2$ instead of $U^{233}O_2$. This increase is caused by the shift in the nuclear criticality

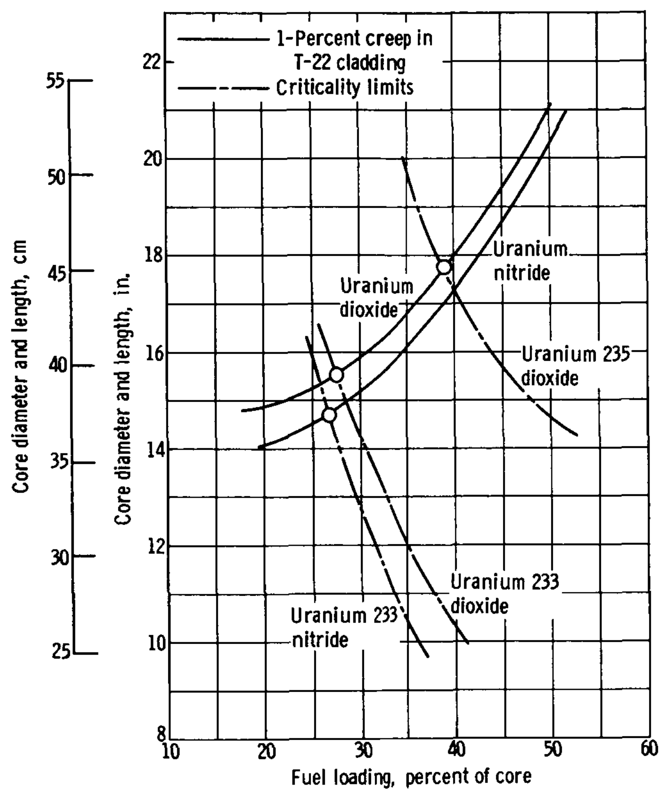


Figure 22. - Effect of fuel material selection on required reactor size. Reactor lifetime, 50 000 hours; thermal power output, 3 megawatts; bulk temperature 2700° R (1500 K); pin diameter, 0.75 inch (19.1 mm); equivalent diameter, 0.125 inch (3.175 mm).

curve associated with the less effective 235 isotope of uranium. The criticality limits (fig. 22) for all three types of fuel were taken from reference 1.

Effect of fission-gas release rate. - The assumed fission product release rate of 0.3-gaseous atoms per fission has been used to establish the behavior of the unvented fuel element. To determine how the required reactor size varies as a function of the fission gas release rate, several additional calculations were performed in which the assumed release rate was increased to values of 0.6, 0.9, and 1.2 gaseous atoms per fission. Figure 23 shows the results of this study. As the fission gas release rate is increased, the required reactor size obviously increases. For the range of fission gas release rates covered in these calculations, the required reactor size increased from 13.5 to 16.5 inches (34.3 to 41.9 cm) for the conditions used. Also shown in figure 23 for comparison is the curve for the vented fuel system (fig. 8).

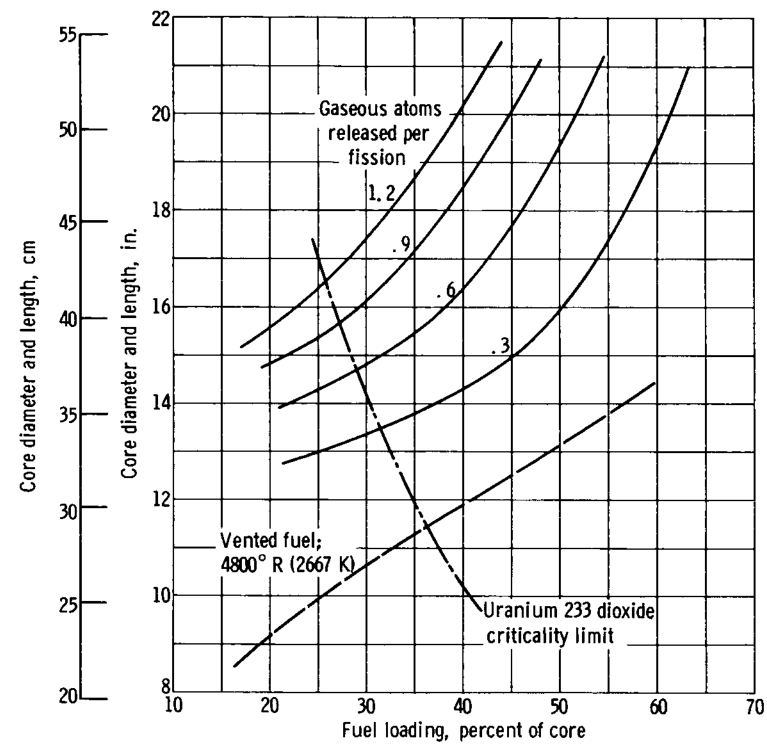


Figure 23. - Effect of gaseous fission product releases on required core core size of unvented pin design. Reactor lifetime, 20 000 hours; thermal power output, 3 megawatts; bulk temperature, 2700° R (1500 K); pin diameter, 0.75 inch (19.1 mm); equivalent diameter, 0.125 inch (3.175 mm).

PARTIALLY VENTED DESIGN

The partially vented design (fig. 3), in which the gaseous-fission products are stored in a chamber that is an integral part of the fuel element, combines some of the features of both the vented and unvented systems. In comparison with the vented system, the partially vented concept eliminates the individual tubes which must be attached to the external storage container. The chief advantage of the partially vented design over the unvented system is that the gas-storage temperature is no longer dependent on fuel temperature, so that the fuel is allowed to attain the maximum temperature of 4800° R (2667 K). This results in a higher allowable heat flux and, therefore, in a smaller reactor core.

Calculations made in the study of the partially vented system were accomplished by varying the clad thickness and void volume (i.e., the length of the integral chamber) to obtain the pressure which satisfies the 1-percent creep limitation (see appendix). Figure 24 shows the resulting core size, excluding the length of the fission gas storage

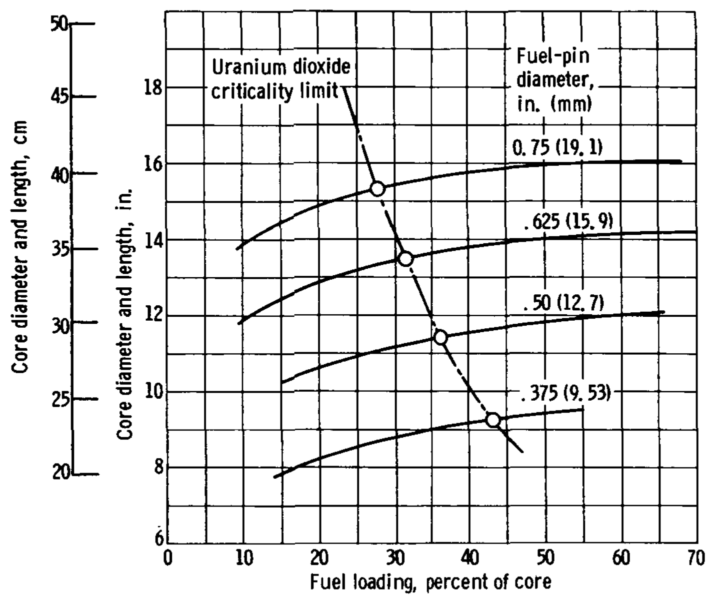


Figure 24. - Effect of pin diameter and available fuel volume on required core size for partially vented concept. Creep limited to 1 percent in T-222 cladding. Maximum fuel temperature, 4800° R (2667 K); diameter of void, 0.175 inch (4.4 mm); reactor lifetime, 20 000 hours; bulk temperature, 2700° R (1500 K); equivalent diameter, 0.125 inch (3.175 mm).

chamber, for a 3-megawatt (thermal), 20 000-hour reactor operating under an assumed set of conditions. The physical size of the reactors are esentially the same as the vented system (fig. 8) since the operating conditions are quite similar; the smaller the fuel pin, the higher the allowable power density and the smaller the required core size. Table II compares the characteristics of the critical cores (fig. 24) for the various diameter fuel pins.

Because of the smaller core size associated with the smaller fuel pins, the average fuel burnup is nearly three times as great (table II) for the 0.375-inch (9.5-mm) diameter pins as it is for the 0.75-inch (19.1-mm) diameter pins. However, because the power level and operating time are the same for all pin sizes, the total number of fissions, and

TABLE II. - COMPARISON OF SIZE REQUIREMENTS OF A PARTIALLY VENTED REACTOR USING VARIOUS DIAMETER FUEL PINS

[Core power, 3 MW_t; core length-to-diameter ratio, 1.0; fuel, $U^{233}O_2$; inlet coolant temperature, 2600° R (1444 K); exit coolant temperature, 2700° R (1500 K); maximum fuel temperature, 4800° R (2667 K); operating time, 20 000 hr; diameter of central void, 0.175 in. (4.4 mm).]

	Fuel-pin diameter, in.; mm			
	0.375; 9.5	0.5; 12.7	0.625; 15.9	0.75; 19.1
Critical fuel loading, percent of core	43	36	32	[b]28
[a]Required clad thickness, in.; mm	[b]0.02; 0.51	0.06; 1.52	0.10; 2.54	0.14; 3.56
Clad fraction, percent of core	[b]14.5	33	44.5	52.5
[c]Coolant fraction, percent of core	26.5	21	17	[b]15
Core void fraction, percent of core	16	10	6.5	[b]4.5
[a]Allowable stress, psi; N/cm^2	7950; 5481	7620; 5254	7440; 5130	[b]7315; 5044
[a]Internal pressure, psi; N/cm^2	[b]880; 607	2040; 1407	2735; 1886	3250; 2241
Maximum heat flux, Btu/(hr)(ft^2); W/cm^2	5×10^5; 158	3.4×10^5; 107	2.5×10^5; 79	[b]1.9×10^5; 60
Average fuel burnup, atom percent	6.2	3.9	2.7	[b]2.1
Core diameter and length, in.; cm	[b]9.3; 23.6	11.5; 29.2	13.5; 34.3	15.3; 38.9
Chamber length, in.; cm	10.3; 26.2	2.8; 7.1	1.4; 3.6	[b]0.8; 2.0
Total length, in.; cm	19.6; 49.8	[b]14.3; 36.3	14.9; 37.8	16.1; 40.9
Core volume, $in.^3$; dm^3	[b]632; 10.4	1194; 19.6	1932; 31.7	2813; 46.1
Total volume, $in.^3$; dm^3	[b]1331; 21.8	1485; 24.3	2133; 35.0	2960; 48.5
Void volume in core, $in.^3$; dm^3	[b]101; 1.7	119; 2.0	126; 2.1	127; 2.1
Fraction of gas retained in core void	[b]12	34	48	61
Fraction of gas held in chamber	88	66	52	[b]39
Void volume in chamber, $in.^3$; dm^3	365; 6.0	119; 2.0	70; 1.1	[b]45; 0.74
Required number of fuel pins	402	375	345	[b]323

[a]Clad thickness optimized to obtain maximum fuel loading.

[b]Minimum value.

[c]At constant hydraulic diameter, 0.125 in. (3.175 mm).

hence the amount of fission gas, that must be accommodated is the same.

With the larger cores, there is a small increase in the available void volume within the core itself; for example, the 0.75-inch (19.1-mm) diameter fuel pin has 26 percent more void within the core than does the 0.375-inch (9.5-mm) fuel pin. This increase in available volume within the core plus the higher internal pressure permitted by the thicker clad of the larger diameter fuel pins (table II) permits a larger fraction of the fission gases to be held within the core.

Sixty-one percent of the fission gas is retained in the core using 0.75-inch (19.1-mm) diameter fuel pins, whereas only 12 percent of the gas can be retained in the much smaller core of the 0.375-inch (9.5-mm) fuel pins. Since the remaining fission gas (i.e., 39 and 88 percent, respectively, for the large and small pins) must be held in the chamber void and since allowable pressure is much greater for the larger pins, the required chamber void volume is necessarily much larger for the small pins. Coupled with the fact that the available volume per unit length of pin varies as the square of the pin diameter, the required length of the chamber for the 0.375-inch (0.5-mm) fuel pins is nearly 13 times as long as for the 0.75-inch (19.1-mm) fuel pins. Figure 25 shows the length of the chamber which is required to satisfy the pressure and stress requirements of the clad.

When the required chamber length (fig. 25) is added to the required core length (fig. 24), the overall or total length of the reactor is obtained (fig. 26(a)). Because of

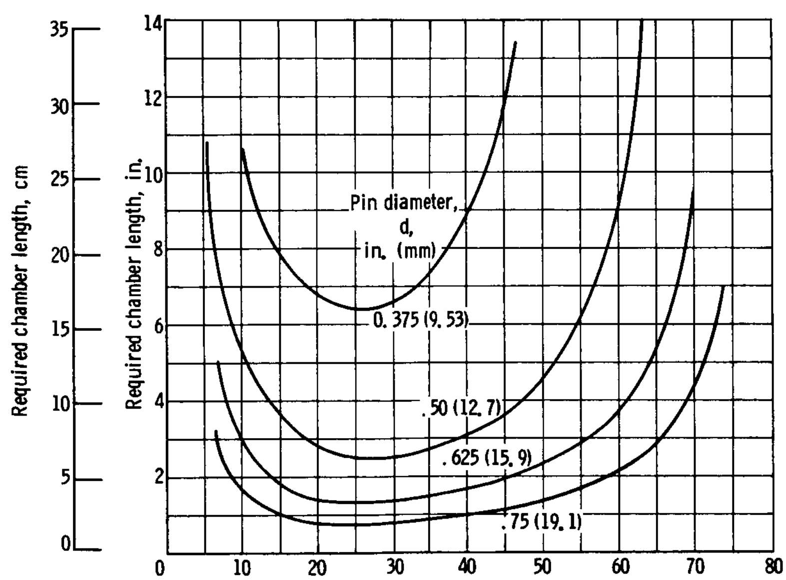
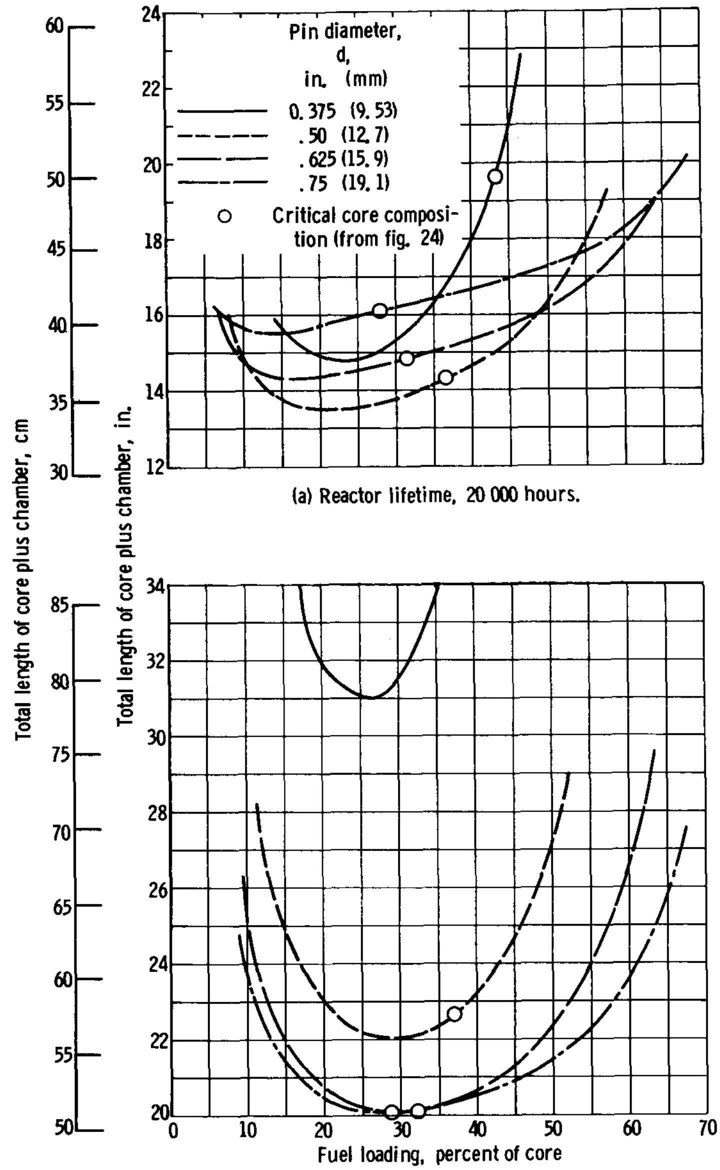


Figure 25. - Effect of pin diameter and available fuel volume on required chamber length of partially vented concept. Creep limited to 1 percent in T-222 cladding. Maximum fuel temperature, 4800° R (2667 K); reactor lifetime, 20 000 hours; bulk temperature, 2700° R (1500 K); equivalent diameter, 0.125 inch (3.175 mm); fuel, uranium dioxide.



(a) Reactor lifetime, 20 000 hours.

(b) Reactor lifetime, 50 000 hours.

Figure 26. - Effect of pin diameter on total length of partially vented concept. Creep limited to 1 percent in T-222 cladding. Maximum fuel temperature, 4800° R (2667 K); diameter of void, 0.175 inch (4.4 mm); bulk temperature, 2700° R (1500 K); equivalent diameter, 0.125 inch (3.175 mm); fuel, uranium dioxide.

the conflicting nature of the core and chamber requirements (i.e., the small fuel pins require a small core but a long chamber), there is an optimum fuel-pin diameter associated with a given set of operating conditions. (Note that the critical core sizes for each of the pin diameters have been obtained from fig. 24.) For the conditions assumed in this example (fig. 26(a)), the 0.50-inch (12.7-mm) diameter fuel pin results in the smallest overall length (14.3 in. or 36.3 cm). However, the minimum volume is not necessarily associated with the minimum overall length as shown in table II. The minimum total volume, as well as the minimum core volume, occurs with the smallest (0.375 in. or 9.5 mm) fuel pin. The choice of using the minimum-volume design or the minimum-length design would depend on the shielding requirements and weight limitation of a particular application.

Changing the operating conditions would, of course, change the fuel-pin diameter required to optimize the overall length. Figure 26(b), for example, shows the overall length of a 50 000-hour reactor. The required core size is essentially the same as in the 20 000-hour case (fig. 24) since all other conditions have been maintained. However, because $2\frac{1}{2}$ times as many fissions have taken place, the chamber length (and chamber void) must be increased to accommodate the gaseous fission products. The resulting overall length of the core and chamber is about 20 inches (50.8 cm). There is very little difference between using 0.625- or 0.75-inch (15.9- or 19.1-mm) diameter fuel pins under the conditions used in this example (fig. 26(b)).

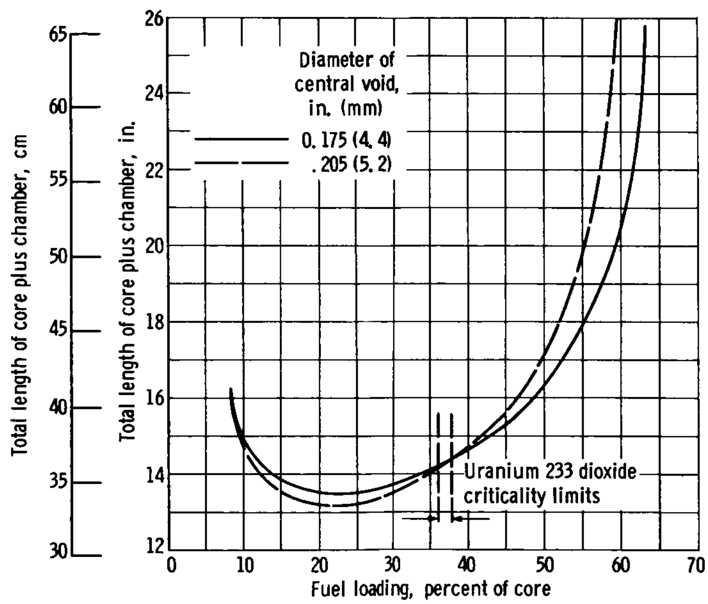


Figure 27. - Effect of vent tube diameter on length of core plus chamber for partially vented concept. Creep limited to 1 percent in T-222 cladding. Maximum fuel temperature, 4800° R (2667 K); inlet coolant temperature, 2600° R (1444 K); bulk temperature, 2700° R (1500 K); reactor lifetime, 20 000 hours; pin diameter, 0.50 inch (12.7 mm); equivalent diameter, 0.125 inch (3.175 mm); fuel, uranium dioxide.

The use of an alternate size vent tube does not have an appreciable effect on the overall length of the reactor for nominal changes in the diameter of the vent tube (fig. 27). The chief reason that the partially vented system requires such a relatively large chamber length is that the temperature to which the gases are cooled (i.e., 2600° R (1444 K), the inlet coolant temperature) is still quite high. In fact, compared with the unvented system where the maximum fuel temperatures (table I) were less than 3600° R (2000 K), the decrease in gas-storage temperature is rather insignificant. It would appear that unless the gas temperature could be reduced to a much lower value, the length of the storage chamber will offset any reduction in the required core size that results from the higher fuel fraction.

The previous calculations on the partially vented concept were performed using a maximum allowable fuel temperature of 4800° R (2667 K). To determine the effect of changing this temperature limit, several additional calculations were made over a wide range of fuel temperatures.

Figure 28 shows the results of this study; the required core diameter and length (L/D = 1), the fission-gas chamber length and the combined or total length are shown as

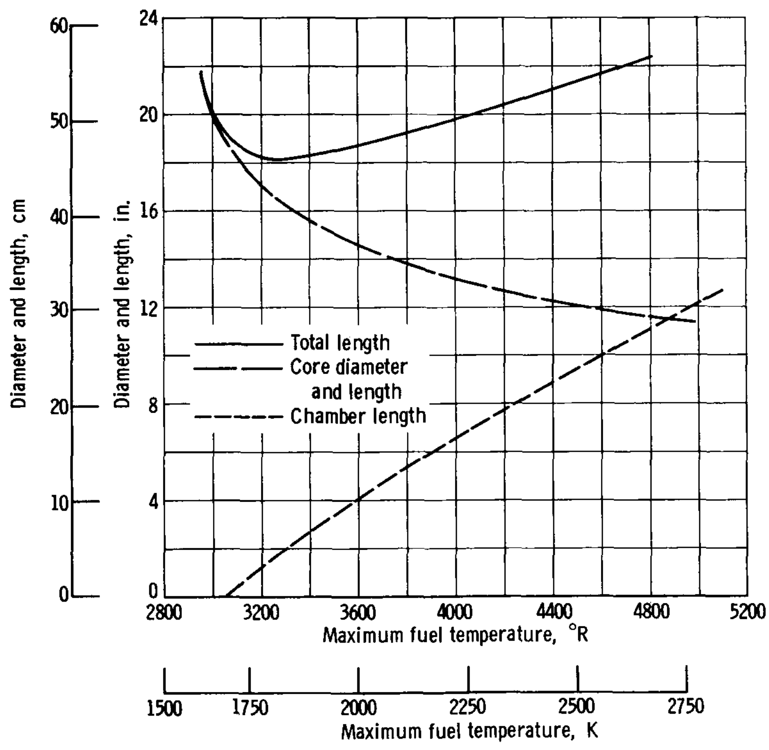


Figure 28. - Effect of allowable maximum fuel temperature on size of partially vented reactor concept. Creep limited to 1 percent in T-222 cladding. Inlet coolant temperature, 2600° R (1444 K); bulk temperature, 2700° R (1500 K); reactor lifetime, 50 000 hours; pin diameter, 0.50 inch (12.7 mm); equivalent diameter, 0.125 inch (3.175 mm); fuel, uranium dioxide.

a function of maximum fuel temperature. As would be expected, as the allowable fuel temperature is reduced, the required core size increases as a result of the reduction in specific power which is required to maintain the lower temperatures. At the same time, the required chamber length decreases because of several related factors:

(1) The larger diameter core increases the number of fuel pins so that the required chamber length necessary to obtain a given chamber volume is reduced.

(2) With the lower power density, the amount of fission gas released per pin is reduced.

(3) The increase in core volume also provides an increase in the void volume within the core.

(4) The reduction in fuel temperature allows more fission gas to be contained with the fuel-element void.

Because of the conflicting behavior of the core length and chamber length, the total length has a minimum value. In the example shown in figure 28, the minimum value occurs at about 18 inches (45.7 cm). However, even though the overall length is minimized at this point, the total volume is not. Table III shows some pertinent dimensions and volumes for the example case (fig. 28). This table shows that the minimum volume actually occurs at the highest fuel temperature rather than at the point of minimum total length. This is due to the fact that, with a reduction in temperature, the core volume increases more rapidly than the required chamber volume decreases. Again, the advantage of choosing the minimum length or the minimum volume design would, of course, depend to a large extent on the proposed application of such a reactor and on whether the

TABLE III. - EFFECT OF ALLOWABLE MAXIMUM FUEL TEMPERATURE ON SIZE

OF PARTIALLY VENTED REACTOR USING URANIUM-DIOXIDE FUEL

Allowable fuel temperature		Required core diameter and length[a]		Required chamber length		Total length		Core volume		Total volume	
^{o}R	K	in.	cm	in.	cm	in.	cm	$in.^3$	cm^3	$in.^3$	cm^3
4800	2667	11.5	29.2	11.0	27.9	22.5	57.1	1180	1.93×10^4	2320	3.80×10^4
4500	2500	12.0	30.5	9.4	23.9	21.4	54.4	1357	2.22	2420	3.97
4000	2222	13.2	33.5	6.6	16.8	19.8	50.3	1805	2.96	2710	4.44
3500	1944	15.2	38.6	3.2	8.1	18.4	46.7	2760	4.52	3340	5.47
3000	1667	20	50.8	0	0	20	50.8	6290	10.3	6290	10.3

[a]Core length-to-diameter ratio of 1.0; pin diameter, 0.5 in. (12.7 mm); equivalent diameter, 0.125 in. (3.175 mm); exit coolant temperature, 2700° R (1500 K); inlet coolent temperature, 2600° R (1444 K); core power, 3 MW; operating lifetime, 50 000 hr.

overall volume or the overall length were more significant in establishing the shielding requirements and weight of the reactor and shield.

MATERIAL STRENGTH REQUIREMENTS

The previous calculations have all been performed by arbitrarily varying the fuel, clad, and void content of the reactor. The required reactor sizes were then established by determining the intersection of the criticality limit with the results obtained.

By altering the calculational procedure somewhat, it is possible to determine what material strength is required to satisfy the 1-percent creep limit for a given set of operating conditions (i.e., power level, time, and temperature).

The procedure followed in calculating the strength requirements for the unvented fuel concept was as follows (see appendix):

(1) The required fuel volume for a given size reactor was determined from the criticality limit (ref. 1); for example, for a reactor size of 12 inches (30.5 cm), a $U^{233}O_2$

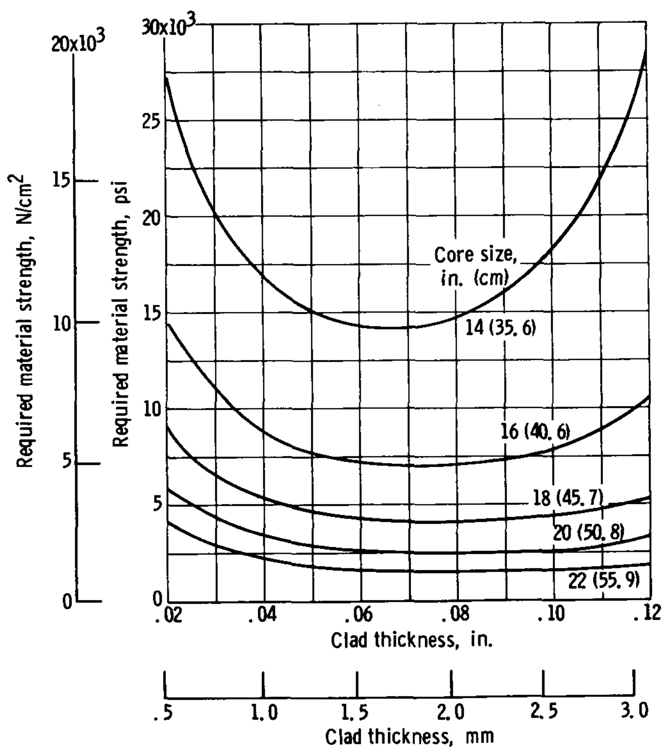


Figure 29. - Effect of core size and clad thickness on required material strength (end-of-life stress level) necessary to limit creep to 1 percent with a linear fission gas release. Thermal power output, 3 megawatts; reactor lifetime, 50 000 hours; pin diameter, 0.75 inch (19.1 mm); clad temperature, 2700° R (1500 K); coolant fraction, 0.20; length-to-diameter ratio, 1.0; fuel, uranium dioxide.

fuel fraction of 35 volume percent is required.

(2) A coolant fraction was chosen (e.g., $C_f = 0.20$).

(3) The remaining volume, in this case 45 percent of the core volume, was assumed to be composed of clad plus void.

(4) For a given clad thickness, the void volume, internal pressure, and stress level were established. This stress level represents the minimum required material strength that would be necessary to produce 1-percent strain in the clad assuming a linear fission gas release.

Figure 29 shows the results of a typical set of calculations made in this manner for a 0.75-inch (19.1-mm) diameter fuel pin. For a given reactor size, the optimum clad thickness (i.e., that thickness which requires the lowest material strength) varies over the range of 0.060 to 0.100 inch (1.52 to 2.54 mm). For all practical purposes, however,

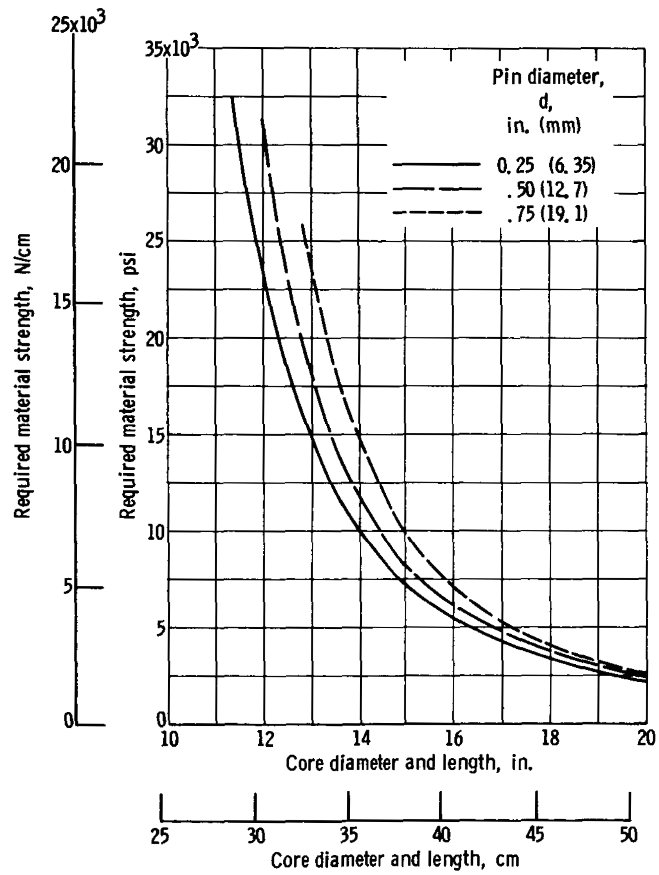


Figure 30. - Effect of fuel pin and core size on required material strength (end-of-life stress level) necessary to limit creep to 1 percent with a linear fission gas release. Thermal power output, 3 megawatts; reactor lifetime, 50 000 hours, pin diameter-to-clad thickness ratio, 10; coolant fraction, 0.20; clad temperature, 2700° R (1500 K); length-to-diameter ratio, 1.0; fuel, uranium dioxide.

the minimum strength occurs around a pin diameter-to-clad thickness ratio (d/t) of 10; this is true regardless of the diameter of the fuel pin.

For the case shown in figure 29, the minimum required strength of a 22-inch (55.9-cm) reactor is on the order of 1500 psi (1034 N/cm^2); at 18 inches (45.7 cm), approximately 4000 psi (2758 N/cm^2); at 14 inches (35.6 cm), the minimum clad strength is nearly 15 000 psi (10 340 N/cm^2); Since the strength of T-222 (fig. 7) is approximately 7500 psi (5170 N/cm^2) at 2700° R (1500 K), the results shown in figure 29 agree quite well with the 16.5-inch (41.9-cm) required core size obtained in the previous calculations (fig. 21).

Figure 30 shows how the use of a smaller pin size changes the required material strength for a given size core. These calculations were all based on a pin diameter-to-clad thickness ratio of 10 and, as in figure 30, were based on the use of $U^{233}O_2$ fuel. Figure 31 shows how the use of other fuel material affects the required material strength.

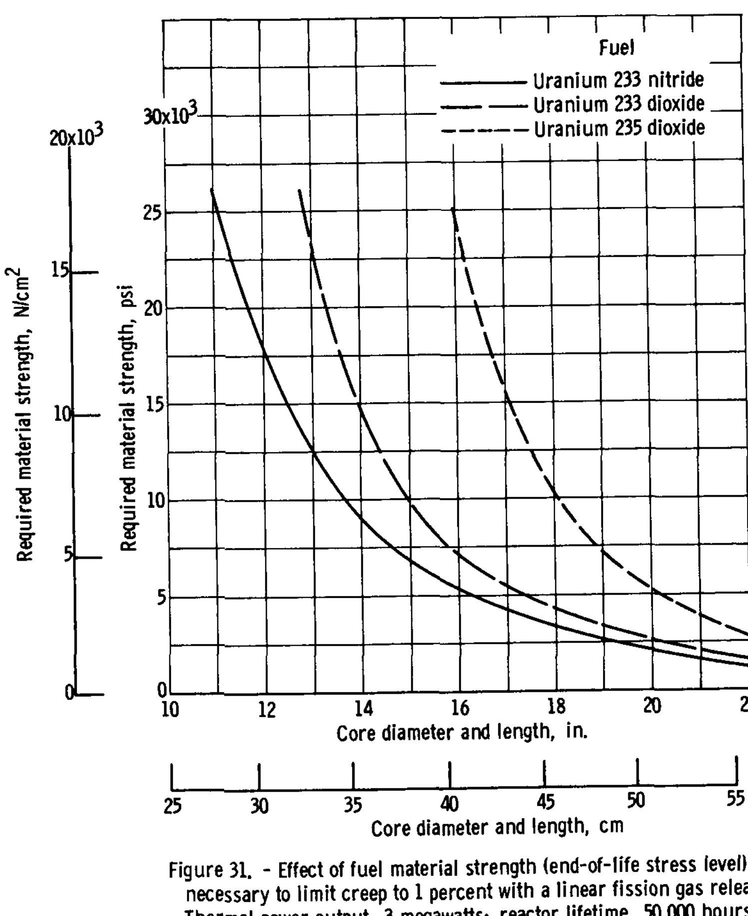


Figure 31. - Effect of fuel material strength (end-of-life stress level) necessary to limit creep to 1 percent with a linear fission gas release. Thermal power output, 3 megawatts; reactor lifetime, 50 000 hours; pin diameter, 0.75 inch (19.1 mm); pin diameter-to-clad thickness ratio, 10; coolant fraction, 0.20; clad temperature, 2700° R (1500 K); length-to-diameter ratio, 1.0.

These results (figs. 30 and 31) agree well with previous results (figs. 19 and 22) obtained with T-222 at this temperature.

Figure 32 shows how operating time and temperature affect the material strength requirements of the clad. The shape of the curves are essentially the same but, as

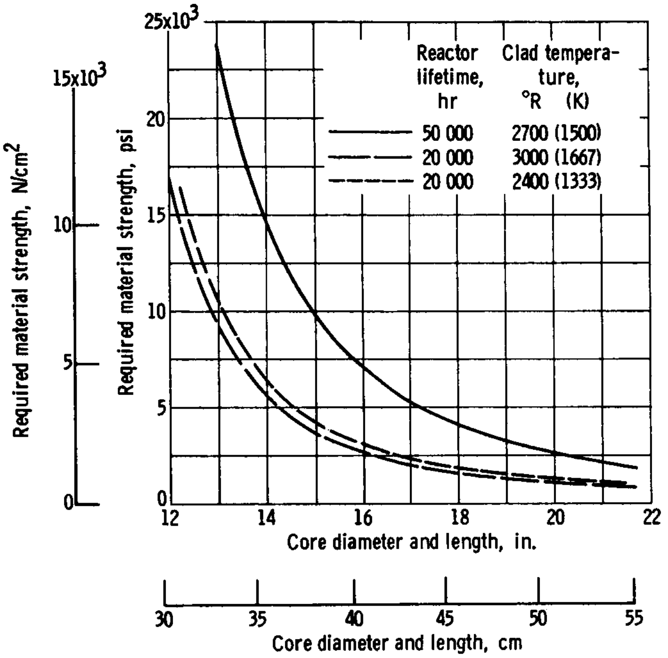


Figure 32. - Effect of operating lifetime and temperature on required material strength (end-of-life stress level) necessary to limit creep to 1 percent with a linear fission gas release. Thermal power output, 3 megawatts; pin diameter, 0.75 inch (19.1 mm); pin diameter-to-clad thickness ratio, 10; coolant fraction, 0.20; length-to-diameter ratio, 1.0; fuel, uranium dioxide.

would be expected, shorter operating times and lower fuel temperatures result in a smaller reactor for a given clad strength. The effect of operating time is much more significant than that of clad temperature. For a clad strength of 7500 psi (5170 N/cm^2), for example, there is nearly a 2.5-inch (6.35-cm) difference in the required reactor size between 20 000 and 50 000 hours of operation. Again, these are about the same results as were previously obtained (fig. 17) for T-222.

Another interesting point that can be seen from the results of these calculations (figs. 29 to 32) is that there are a minimum usable strength and a maximum advantageous strength for this particular application. This is because of the asymptotic behavior of these results. At the low material strengths, that is, less than 5000 psi (3450 N/cm^2), a small reduction in material strength causes a disproportionate increase in the required reactor size (fig. 30). Therefore, the reactor is severely penalized if the material strength drops below this minimum usable value.

At the other end of the scale, when the material strength is greater than 15 000 psi (10 340 N/cm^2), only a very small decrease in reactor size accompanies a two- or three-fold increase in the material strength once the maximum advantageous strength has been achieved. Although it is quite difficult to establish the exact maxima and minima strengths for all possible combinations of reactor parameters that could be considered, it would appear that at the temperatures of interest a clad with a strength of from 5000 to 15 000 psi (3450 to 10 340 N/cm^2) would be reasonable. However, if it is assumed that the criticality limits do not shift appreciably with the use of other cladding materials, it is possible to estimate the size advantage of using a stronger clad material. At 2700° R (1500 K), for example, the strength of W-25Re (fig. 7) for 50 000 hours of operation is over 25 000 psi (17 240 N/cm^2) so that core sizes of around 11 to 13 inches (27.9 to 33) would be possible with U^{233} fuel (fig. 32) providing the high burnups associated with this size reactor (figs. 4 and 5) do not cause excessive fuel swelling or adversely affect nuclear criticality. This is approximately the same size cores as can be achieved with the vented system (fig. 8) which indicates that, if the cladding is sufficiently strong, there is no real advantage in using the more complex vented fuel-pin design.

EFFECT OF POOR THERMAL BOND

One of the assumptions made in the analysis was that a good thermal bond existed between the clad and fuel. The premise behind this assumption is that at high temperatures the fuel evaporates and then condenses on the cooler clad surface. All the microscopic grooves and scratches are thereby filled, and intimate contact between the clad and fuel is achieved. If this condition does not occur, an additional source of error in the temperature calculations will occur. The magnitude of the error resulting from this assumption can be approximated by considering an alternate model in which the contact between clad and fuel is not as intimate as in the model used in the analysis.

Initially, the fuel and clad can be sized to take advantage of the larger thermal coefficient of expansion of the fuel so that, under operating conditions, the two materials are in contact. There are, however, uncertainties in establishing the required differential expansion coefficients, uncertainties in the temperature calculations, and certain manufacturing tolerances which also could result in a mismatch in the thermal expansion between clad and fuel. In addition, there is the problem of radiation-induced fuel swelling, which can also increase the contact pressure of the fuel on the clad.

Because of the many uncertainties in predicting the exact contact pressure between the fuel and clad, the prediction of the contact coefficient h_c is quite difficult. However, several experimental measurements of contact coefficients have been made. For example, Arrighi, Mustacchi, and Zanella (ref. 27) have made measurements of fuel-clad contact resistance under a variety of conditions and, in the region of hot contact, the steady-state contact coefficients varied from 1000 to 2000 Btu per hour per square foot per ^{o}R (0.57 to 1.14 $W/(cm^2)(K)$). Similar results have been obtained in other experiments (refs. 28 and 29).

The additional temperature rise (ΔT_c) associated with the contact resistance between fuel and clad is given by

$$\Delta T_c = \frac{F_{r,z}\varphi b}{h_c a} \tag{8}$$

where the symbols are those used in the previous heat-transfer expression (eq. (6)). The subscript c denotes the contact value. For a typical case, such as for the unvented 0.75-inch (19.1 mm) diameter fuel pin listed in table I, the additional temperature rise (ΔT_c) would be 170^{o} R (94 K) (i.e., assuming the 1000 Btu/(hr)(ft^2)(^{o}R) or 0.57 W/(cm)(K) contact coefficient). The maximum internal temperature for this case would then be 3740^{o} R (2078 K) which is still well below the assumed allowable temperature of 4800^{o} R (2667 K). Even for the vented and partially vented cases, an additional 170^{o} to 250^{o} R (94 to 139 K) temperature rise should not result in any problem because the maximum fuel temperature would still be below the melting point of the fuel.

Because of the difficulties associated with accurately predicting the thermal expansion and fuel swelling effects, it is possible that the initial sizing of fuel and clad would be such that during normal operation there would be no contact. In this case, the temperature rise between clad and fuel would be given by

$$\Delta T_g = \frac{b\varphi}{K_g} \ln\left(1 + \frac{g}{a}\right) \tag{9}$$

where g is the gap thickness and the subscript g refers to the conditions in the gap. This gap would be filled with a mixture of the fission gases and the inert gas with which the fuel element had been filled during manufacturing. Assuming that the thermal conductivity of the mixtures is 0.04 Btu per hour per foot per ^{o}R (6.9×10^{-4} W/(cm)(K)) and that the width of the operating gap is 0.001 inch (0.025 mm), the temperature rise across the gap for the example of the unvented fuel pin used previously is 435^{o} R (240 K). The maximum temperature for this 0.75-inch (1911 mm) fuel pin is then 4005^{o} R (2225 K), which is still considerably lower than the melting temperature of UO_2 fuel.

Therefore, if the assumption of a good thermal bond is found to be invalid, the temperatures associated with an alternate model (i.e., either a contact coefficient or a gas gap) are still tolerable. Some increase in the size of the reactor will have to be made, however, to compensate for the increase in pressure stresses due to the somewhat higher fuel temperature.

CONCLUDING REMARKS

As a result of the preliminary study on the size requirements of a fast-spectrum, liquid-metal cooled reactor utilizing pin-type fuel elements, several areas of uncertainty were identified in which additional information is required to perform an accurate analysis of the vented and unvented fuel elements considered. These areas were (1) fission gas release rates and determination of the fission gas pressure, (2) effect of high fuel burnups on fuel swelling and nuclear control, (3) long-term, high-temperature creep behavior of potential cladding materials, (4) circumferential heat transfer in closely spaced rods, and (5) the stress and creep behavior in a fuel rod having a nonuniform circumferential temperature distribution.

Although additional information in these categories is required for a complete fuel-pin analysis, the general characteristics of the various concepts have been demonstrated in this study. Several of these points are summarized as follows:

1. Required sizes for a 3-megawatt-thermal vented reactor are on the order of 10 to 13 inches (25.4 to 33 cm) using a reasonable size vent tube and uranium 233 dioxide fuel operating at 4800° R (2667 K). The minimum usable reactor size will probably be determined by the maximum fuel depletion that can be tolerated from fuel swelling and nuclear control consideration. Although the vented fuel system is more complex to construct, the resulting reactor sizes are generally smaller than those for the unvented fuel design. This size advantage decreases as the allowable fuel-temperature limit is reduced.

2. The limiting factor in the unvented fuel pin is the strength of the cladding. Using T-222 cladding, the required size of a 3-megawatt (thermal), uranium 233 dioxide fueled, unvented core varies from 13 to 17 inches (33 to 43.2 cm) over a range of operating conditions (i.e., for variations in operating time, temperature, pin sizes, and pin spacing). If a stronger clad material could be used (e.g., W - 25Re), the required size of the unvented core would be approximately the same as the vented core.

3. The partially vented system has no appreciable size advantages over either the vented or unvented systems. Although the core size is smaller, the combined length of the core and fission gas storage chamber is larger than the unvented design. For a given set of operating conditions, there is an optimum fuel-pin diameter which will result in

the minimum overall length (i.e., length of core plus chamber). However, the minimum volume is associated with the smallest diameter fuel pin. The choice of using the minimum volume design or minimum overall length design depends on the shielding requirements and weight limitation of a particular application.

4. The use of closely spaced small-diameter fuel pins generally results in the smallest core size. However, the number of fuel pins required for a given size reactor is inversely proportional to the square of the pin diameter. Also, the fabrication and assembly tolerances will be much more restrictive for small fuel pins with close spacing, and circumferential heat-transfer variation will occur. Conversely, if the fuel pins are too large, the fuel temperature and stress problems are more acute and larger core sizes are required. The choice of one pin size over another should be based on a reasonable compromise between these factors.

5. Using the criticality limits established in reference 1 for this general type of reactor, it was found that the use of uranium 233 instead of uranium 235 fuel or the use of uranium nitride rather than uranium dioxide fuel will decrease the required reactor size. For an unvented system the use of uranium 233 dioxide instead of uranium 235 dioxide reduces the required reactor diameter and length by about 3 inches (7.6 cm); the use of uranium nitride saves an additional inch (2.54 cm) in core size over uranium dioxide fuel. Although it would appear advantageous to select the uranium 233 nitride fuel based only on the core size requirements, there are other factors to consider in the the selection of the fuel. High-level gamma radiation from uranium 233 makes it difficult to fabricate fuel elements without special remote handling equipment; also, uranium nitride fuel is not compatable with the tantalum cladding (T-222) considered in this study.

6. The use of a cladding material which is stronger than T-222 at the temperatures of interest, reduces the required core size in the unvented concept. However, there is a point beyond which only a small reduction in core size can be achieved for large increases in material strength. Conversely, there is a point below which a small reduction in material strength causes a disproportionate increase in the required core size. For the conditions considered in this study, the range of useful material strengths was found to be between 5000 to 15 000 psi (3450 and 10 350 N/cm^2).

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, April 3, 1968,
120-27-06-17-22.

APPENDIX - CALCULATIONAL PROCEDURES FOR DETERMINING REQUIRED CORE SIZE FOR REACTORS IN WHICH GASEOUS FISSION PRODUCT CONTAINMENT IS CONSIDERED

The purpose of this appendix is to outline the step-by-step calculational procedures used in this study. Flow charts (figs. 33 to 36) showing the methods used to obtain the results discussed in the test are given. In order to better demonstrate the effect of the parameters studied, many of the results plotted in the text represent intermediate steps rather than the final solution defined by the charts.

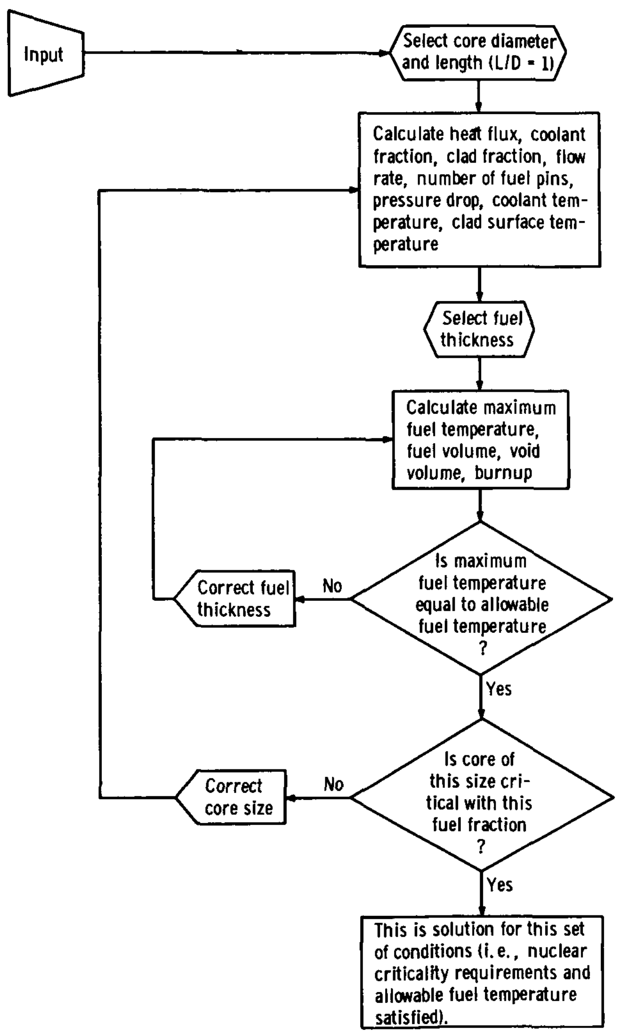


Figure 33. - Calculations of vented fuel pin. Input: Reactor power, fuel pin diameter and spacing, coolant inlet (or exit) temperature, coolant temperature rise, spatial power distribution, clad and fuel properties, maximum allowable fuel temperature, minimum clad thickness, reactor lifetime, nuclear criticality requirements.

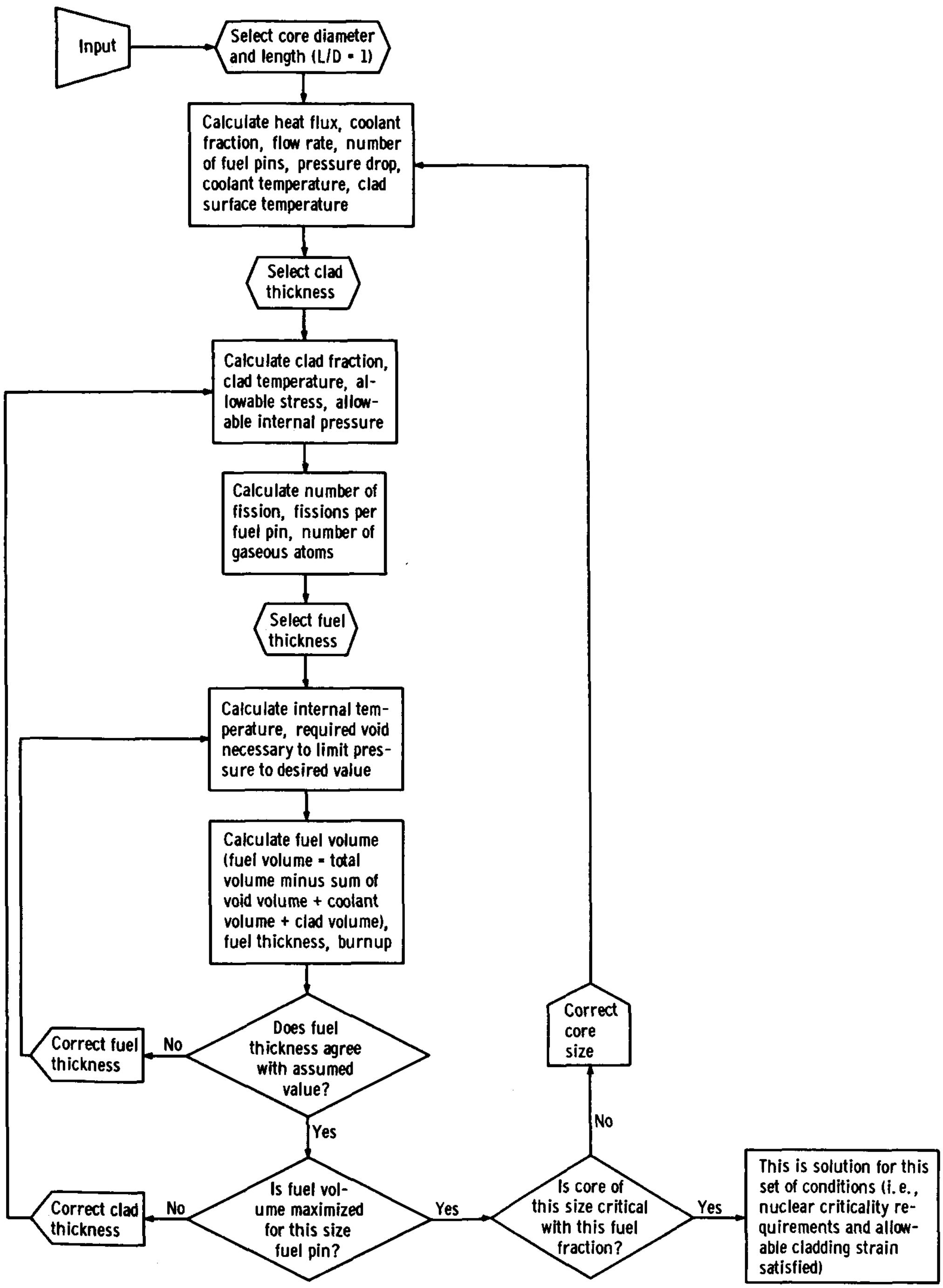


Figure 34. - Calculations of unvented fuel pin. Input: Reactor power, fuel pin diameter and spacing, coolant inlet (or exit) temperature, coolant temperature rise, spatial power distributions, clad and fuel properties, reactor lifetime, nuclear criticality requirements.

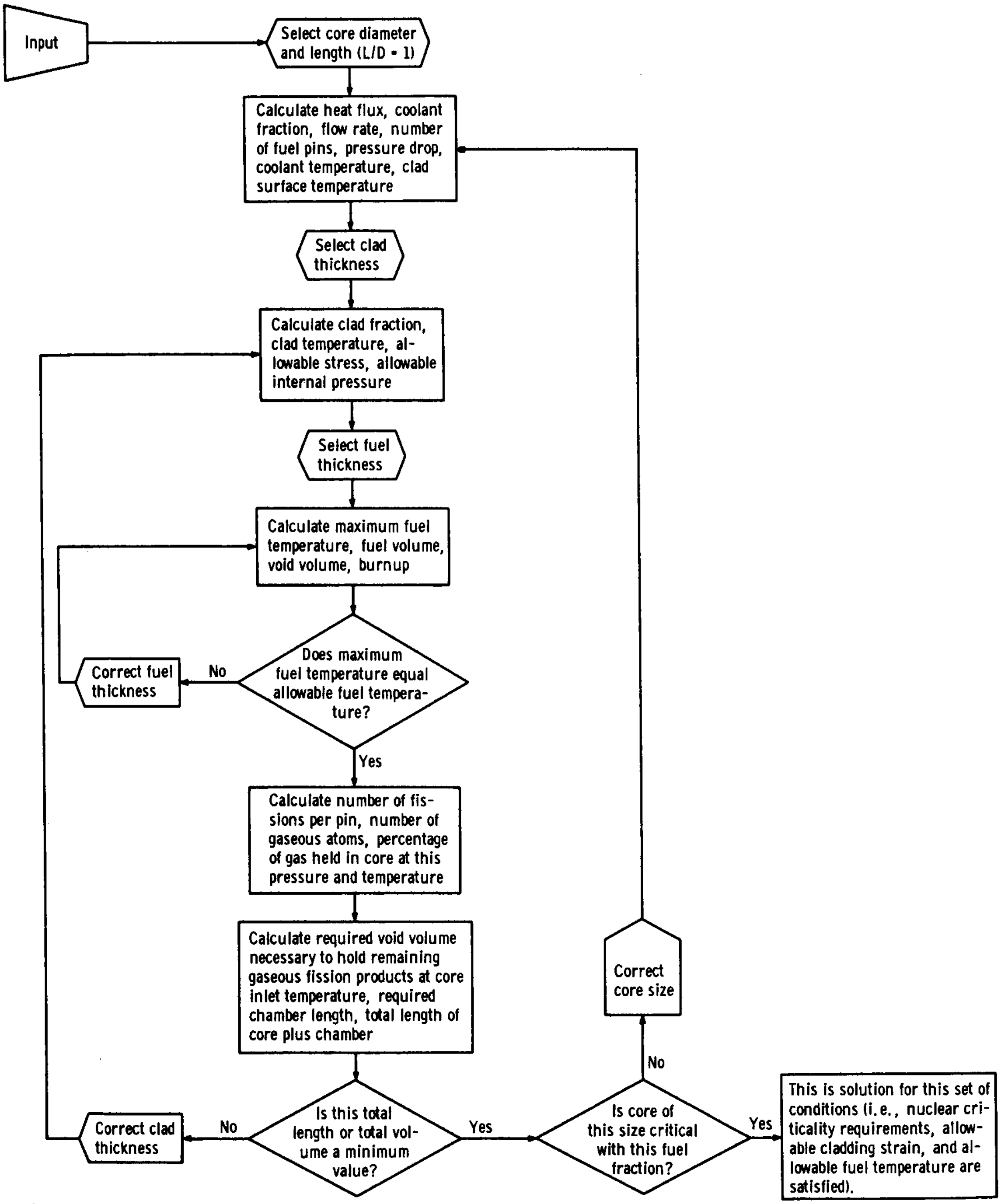


Figure 35. - Calculations of partially vented fuel pin. Input: Reactor power, fuel pin diameter and spacing, coolant inlet (or exit) temperature, coolant temperature rise, spatial power distribution, maximum allowable fuel temperature, clad and fuel properties, reactor lifetime, nuclear criticality requirements.

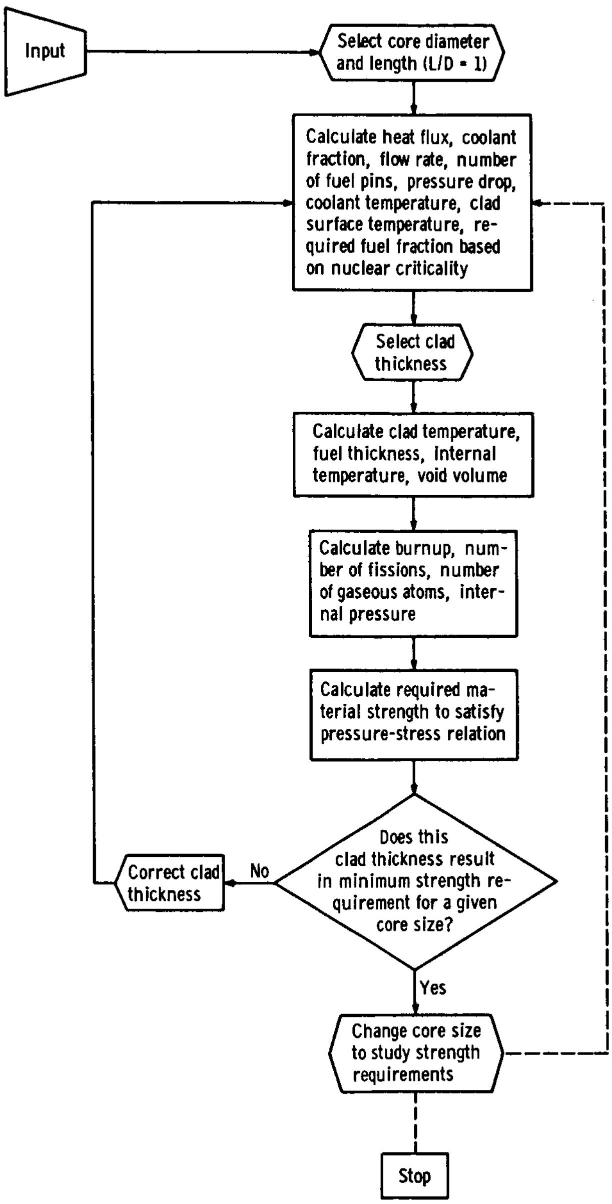


Figure 36. - Calculations of required material strength for unvented fuel pin. Input: Reactor power, fuel pin diameter and spacing, coolant inlet (or exit) temperature, coolant temperature rise, spatial power distribution, fuel properties, reactor lifetime, nuclear criticality requirements.

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